QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

1. 001AA2.03 076/NEW//H/3/4.5/4.8/1/

Given the following conditions:

- Unit 1 is at 75% power and ramping up at 0.3% per minute.
- The crew notes that rods are withdrawing at maximum speed.
- The OATC places rods in MANUAL and reports that rods have stopped moving.

The following alarms are LIT:

- 1A-D1, ROD CONTROL URGENT FAILURE.
- 1A-G2, RPI ROD BOT ROD DROP.
- 1D-E4, NIS PWR RGE HI FLUX RATE RX TRIP.

The OATC reports that Reactor Trip breakers are closed.

Which ONE of the following identifies the procedure flowpath to mitigate this event, **AND** includes the action required if the crew is unable to verify adequate negative reactivity insertion per 1-FR-S.1, Response to Nuclear Power Generation/ATWS?

A. Enter 1-FR-S.1 directly;
manually align Charging Pump suction to RWST and inject the BIT.

B. Enter 1-FR-S.1 directly;
initiate boration at greater than 10 gpm with the blender in BORATE mode.

C. Enter 1-E-0, Reactor Trip or Safety Injection, then transition to 1-FR-S.1;
manually align Charging Pump suction to RWST and inject the BIT.

D. Enter 1-E-0, Reactor Trip or Safety Injection, then transition to 1-FR-S.1;
initiate boration at greater than 10 gpm with the blender in BORATE mode.

a. Incorrect. Plausible since the first 2 steps of both procedures are IOAs and are identical, however FR-S.1 is NOT a direct entry procedure. Second part is correct, this method has consequences, but they are outweighed by the potential greater consequences of not achieving subcriticality in as rapid a manner as possible.

b. Incorrect. First part incorrect but plausible as noted above; second part is incorrect but plausible as this is an alternative offered by ES-0.1. As noted above injecting the BIT has consequences, and being aware of this alternative the candidate who lacks detailed knowledge of the subsequent actions of FR-S.1 may default to this choice.

c. Correct. First part is correct as noted above FR-S.1 is entered only from E-0 or F-0 (E-0, step 1 RNO for these plant conditions); second part is correct as discussed in distractor a.

d. Incorrect. First part is correct as discussed in answer c. Second part is incorrect as discussed in distractor b.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Continuous Rod Withdrawal

Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal :
(CFR: 43.5 / 45.13)

Proper actions to be taken if automatic safety functions have not taken place

Tier: 1
Group: 2

Technical Reference: OP-AP-104, EOP E-0, EOP FR-S.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

Nuclear Fleet

Dominion®

Administrative Procedure

Title: Emergency and Abnormal Operating Procedures

Procedure Number	Revision Number	Effective Date and Approvals On File
OP-AP-104	2	

Revision Summary

Revised to clarify the reading of NOTEs and CAUTIONs:

- Updated 3.4.1.e (EOPs) as follows:
  - OLD - "During performance of all EOPs, the US shall read aloud all applicable NOTEs and CAUTIONs in their entirety the first time they are encountered. When encountered again, the US may read or paraphrase the applicable NOTEs and CAUTIONs. The ROs will paraphrase the Caution or Note and must not respond: understand the NOTEs and CAUTIONs, as the paraphrase. The US will complete the 3-way communication. If the NOTE or CAUTION pertains only to the US or is not applicable at the time, the US is not required to read aloud."
  - NEW - "During performance of all EOPs, the US shall read aloud all applicable NOTEs and CAUTIONs in their entirety the first time they are encountered. When encountered again, the US may read or paraphrase the applicable NOTEs and CAUTIONs. The ROs will paraphrase or acknowledge the NOTE or CAUTION. If the NOTE or CAUTION pertains only to the US or is not applicable at the time, the US is not required to read aloud."
- Updated 3.7.2 (fourth bullet) (AOPs) as follows:
  - OLD - "A NOTE or CAUTION may be directed to applicable crew members; in such cases, repeat-back of the paraphrased NOTE or CAUTION is expected"
  - NEW - "A NOTE or CAUTION may be directed to applicable crew members; in such cases, the applicable crew member will paraphrase or acknowledge the NOTE or CAUTION"

Functional Area Manager: Manager Nuclear Operations

INFORMATION USE

3.1.5 **REPORT** significant or relevant failures of components to the Unit Supervisor or Shift Manager.

NOTE: It is important that all members of the team communicate any concern or objection, including reasons for the concern or objections, in order to resolve problems and enhance decision making.

3.1.6 **COMMUNICATE** any concern or objection to the intended course of action in a timely fashion.

3.2 Entry Into Emergency Operating Procedures - General

Procedure User

3.2.1 **BEGIN** usage with initial emergency procedure for, Reactor Trip or Safety Injection, or loss of all AC power.

Kewaunee	E-0; ECA 0.0
Millstone Unit 2	EOP 2525
Millstone Unit 3	E-0; ECA 0.0
North Anna	E-0; ECA 0.0
Surry	E-0; ECA 0.0

Correct answer
See distractor
analysis for
discussion of
alternatives.

Dominion

NORTH ANNA POWER STATION

FUNCTION RESTORATION PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-FR-S.1	RESPONSE TO NUCLEAR POWER GENERATION/ATWS (WITH FOUR ATTACHMENTS)	16
		PAGE 1 of 12

PURPOSE

To provide instructions for adding negative reactivity to a core that is observed to be critical when expected to be shutdown.

ENTRY CONDITIONS

This procedure is entered from:

- 1-E-0, REACTOR TRIP OR SAFETY INJECTION, or
- Red or Orange terminus of the SUBCRITICALITY CSF STATUS TREE.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: RCPs should not be tripped with Reactor power greater than 5%.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]___	VERIFY REACTOR TRIP:	☐ Verify or place control rods in AUTO.
	☐ a) Manually Trip Reactor	
	b) Check the following:	
	☐ • Reactor Trip and Bypass Breakers - OPEN	
	☐ • Rod Bottom Lights - LIT	
	☐ • Neutron flux - DECREASING	
[2]___	VERIFY TURBINE TRIP:	
	☐ a) Manually Trip Turbine	
	☐ b) Verify all Turbine Stop Valves - CLOSED	☐ b) Put both EHC Pumps in PTL. ☐ <u>IF</u> Turbine is still <u>NOT</u> tripped, <u>THEN</u> manually run back Turbine. ☐ <u>IF</u> Turbine cannot be run back, <u>THEN</u> close MSTVs and Bypass Valves.
	☐ c) Reset Reheaters	
	☐ d) Verify Generator Output Breaker - OPEN	☐ d) <u>IF</u> Generator Output Breaker does <u>NOT</u> open after 30 seconds, <u>THEN</u> manually open G-12 <u>AND</u> Exciter Field Breaker.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3]___	VERIFY CONTROL RODS - INSERTING IN AUTO AT GREATER THAN 48 STEPS/ MINUTE	☐ Manually insert control rods.
4.___	CHECK ALL AFW PUMPS - RUNNING	☐ Manually start pumps.
5.___	**INITIATE EMERGENCY BORATION OF RCS:**	
	☐ a) Verify at least one Charging Pump - RUNNING	☐ a) Start Charging Pumps as necessary.
	b) Emergency borate:	
	☐ 1) Put Boric Acid Transfer Pump in FAST	
	☐ 2) Open 1-CH-MOV-1350, Emergency Borate Valve	

(STEP 5 CONTINUED ON NEXT PAGE)

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	INITIATE EMERGENCY BORATION OF RCS: (Continued)	Answer, see distracter analysis for discussion of alternatives.
	c) Verify adequate negative reactivity insertion:	c) Inject the BIT:
	☐ • Emergency boration - FLOW INDICATED AND	1) Open Charging Pump Suction From RWST Isolation Valves: ☐ • 1-CH-MOV-1115B ☐ • 1-CH-MOV-1115D
	☐ • Control Rods - MOVING IN OR FULLY INSERTED AND	2) Close Charging Pump Suction From VCT Isolation Valves: ☐ • 1-CH-MOV-1115C ☐ • 1-CH-MOV-1115E
	☐ • Neutron flux - DECREASING	3) Close BIT Recirc Valves: ☐ • 1-SI-TV-1884A ☐ • 1-SI-TV-1884B ☐ • 1-SI-TV-1884C

(STEP 5 CONTINUED ON NEXT PAGE)

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	INITIATE EMERGENCY BORATION OF RCS: (Continued)	
		4) Open BIT Outlet Valves: ☐ • 1-SI-MOV-1867C ☐ • 1-SI-MOV-1867D
		5) Open BIT Inlet Valves: ☐ • 1-SI-MOV-1867A ☐ • 1-SI-MOV-1867B
		6) Close Letdown Valves: ☐ • 1-CH-HCV-1200A ☐ • 1-CH-HCV-1200B ☐ • 1-CH-HCV-1200C ☐ • 1-CH-LCV-1460A ☐ • 1-CH-LCV-1460B
		7) Close Normal Charging Line Isolation Valves: ☐ • 1-CH-MOV-1289A ☐ • 1-CH-MOV-1289B
	☐ d) Check PRZR pressure - LESS THAN 2335 PSIG	☐ d) Verify PRZR PORVs and Block Valves are open. ☐ <u>IF</u> <u>NOT</u>, <u>THEN</u> open PRZR PORVs and Block Valves as necessary until PRZR pressure is less than 2335 psig.

2. 006G2.2.39 077/NEW//H/4/3.9/4.5/2/

Unit 1 is at 100% power.

Which ONE of the following places the Unit in a one hour Technical Specification required action, **AND** includes the Bases?

A. BIT boron concentration is 10,950 ppm;
ensures adequate negative reactivity insertion in the injection phase of a design-basis LOCA.

B. BIT boron concentration is 10,950 ppm;
ensures adequate negative reactivity insertion in the injection phase of a design-basis MSLB.

C. Service Water pump 1-SW-P-1A trips;
ensures the system can perform its design function during a DBA with NO additional failures.

D. Service Water pump 1-SW-P-1A trips;
ensures the system can perform its design function during a DBA with ONE additional failure.

a. Incorrect. First part is correct since this is below the minimum value given in the TS surveillance requirements. Second part is incorrect but plausible; the BIT does insert negative reactivity, however the Tech Spec bases notes that the BIT is not credited for immediate boration in the LOCA analysis, but rather for post-LOCA considerations(see TS 3.5.6 bases). The candidate who lacks detailed knowledge of the TS bases may not be aware of this subtle but important difference.

b. Correct. First part is correct as discussed in distractor a. Second part is also correct, the TS bases notes that for the MSLB the BIT is the primary mechanism for inserting boron in the core.

c. Incorrect. First part is incorrect but plausible since the action for 2 pumps is 1 hour; it is the same action for one pump but for that case it is 72 hours. Second part is also plausible since it would be correct prior to the required action being completed (see TS 3.7.8 bases).

d. Incorrect. First part is incorrect but plausible as discussed in distractor c. Second Second part is also plausible since it would be correct after the required action is complete.

Emergency Core Cooling System

Knowledge of less than or equal to one hour Technical Specification action statements for systems.
(CFR: 41.7 / 41.10 / 43.2 / 45.13)

Tier: 2
Group: 1

Technical Reference: TS 3.5.6 and Bases, TS 3.7.8 & bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.6 Boron Injection Tank (BIT)

LCO 3.5.6 The BIT shall be OPERABLE. (see next pg for surv. reqmt)

APPLICABILITY: MODES 1, 2, and 3.

answer

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. BIT inoperable.	A.1 Restore BIT to OPERABLE status.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Borate to an SDM within the limit provided in the COLR. <u>AND</u> B.3 Restore BIT to OPERABLE status.	6 hours 6 hours 7 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 4.	12 hours

See attached TS Bases for other part of answer

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify BIT borated water temperature is ≥ 115°F.	24 hours
SR 3.5.6.2	Verify BIT borated water volume is ≥ 900 gallons.	7 days
SR 3.5.6.3	Verify BIT boron concentration is ≥ 12,950 ppm and ≤ 15,750 ppm.	7 days

not met ∴ BIT INOP

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.6 Boron Injection Tank (BIT)

BASES

BACKGROUND

The BIT is the primary means of quickly introducing negative reactivity into the Reactor Coolant System (RCS) on a safety injection (SI) signal.

The main flow path through the Boron Injection Tank is from the discharge of the High Head Safety Injection (HHSI) pumps through lines equipped with a flow element and two valves in parallel that open on an SI signal. The valves can be operated from the main control board. The valves and flow elements have main control board indications. Downstream of these valves, the flow enters the BIT (Ref. 1).

The BIT is a stainless steel clad tank containing concentrated boric acid. Two trains of strip heaters are mounted on the tank to keep the temperature of the boric acid solution above the precipitation point. The strip heaters are controlled by temperature elements located near the bottom of the BIT. The temperature elements also activate High and Low temperature alarms in the Control Room. In addition to the strip heaters on the BIT, there is a recirculation system with a heat tracing system, including the piping section between the motor operated isolation valves, which further ensures that the boric acid stays in solution. The entire contents of the BIT are injected when required; thus, the contained and deliverable volumes are the same.

During normal operation, a boric acid transfer pump provides recirculation between the boric acid tank and the BIT. On receipt of an SI signal, the recirculation line valves close. Flow to the BIT is then supplied from the HHSI pumps. The solution of the BIT is injected into the RCS through the RCS cold legs.

APPLICABLE SAFETY ANALYSES

During a main steam line break (MSLB) or loss of coolant accident (LOCA), the BIT provides an immediate source of concentrated boric acid that quickly introduces negative reactivity into the RCS.

(continued)

See next pg.

makes the distractor incorrect

BASES

APPLICABLE SAFETY ANALYSES (continued)

The contents of the BIT are not credited for core cooling or immediate boration in the LOCA analysis, but are for post LOCA recovery. The BIT maximum boron concentration of 15,750 ppm is used to determine the minimum time for hot leg recirculation switchover. The minimum boron concentration of 12,950 ppm is used to determine the minimum mixed mean sump boron concentration for post LOCA shutdown requirements.

For the MSLB, the BIT is the primary mechanism for injecting boron into the core to counteract the positive increases in reactivity caused by an RCS cooldown. The MSLB core response analysis conservatively assumes a 0 ppm minimum boron concentration of the BIT, which also affects the departure from nucleate boiling design analysis. The MSLB containment response analysis conservatively assumes a 2000 ppm minimum boron concentration of the BIT. Reference to the LOCA and MSLB analyses is used to assess changes to the BIT to evaluate their effect on the acceptance limits contained in these analyses.

Answer

The minimum temperature limit of 115°F for the BIT ensures that the solution does not reach the boric acid precipitation point. The temperature of the solution is monitored and alarmed on the main control board.

The BIT boron concentration limits are established to ensure that the core remains subcritical during post LOCA recovery. The BIT will counteract any positive increases in reactivity caused by an RCS cooldown.

The BIT water volume of 900 gallons is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to shut down the core following an MSLB, to determine the hot leg recirculation switchover time, and to safeguard against boron precipitation.

The BIT satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum requirements for contained volume, boron concentration, and temperature of the BIT inventory. This ensures that an adequate supply of borated water is available in the event of a LOCA or MSLB to maintain the reactor subcritical following these accidents.

(continued)

BASES

LCO (continued)	To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, and temperature must be met.
APPLICABILITY	In MODES 1, 2, and 3, the BIT OPERABILITY requirements are consistent with those of LCO 3.5.2, "ECCS—Operating." In MODES 4, 5, and 6, the respective accidents are less severe, so the BIT is not required in these lower MODES.

ACTIONS

A.1

If the required volume is not present in the BIT, both the hot leg recirculation switchover time analysis and the boron precipitation analysis may not be correct. Under these conditions, prompt action must be taken to restore the volume to above its required limit to declare the tank OPERABLE, or the unit must be placed in a MODE in which the BIT is not required.

The BIT boron concentration is considered in the hot leg recirculation switchover time analysis, the boron precipitation analysis, and may effect the reactivity analysis for an MSLB. If the concentration were not within the required limits, these analyses could not be relied on. Under these conditions, prompt action must be taken to restore the concentration to within its required limits, or the unit must be placed in a MODE in which the BIT is not required.

The BIT temperature limit is established to ensure that the solution does not reach the boric acid crystallization point. If the temperature of the solution drops below the minimum, prompt action must be taken to raise the temperature and declare the tank OPERABLE, or the unit must be placed in a MODE in which the BIT is not required.

The 1 hour Completion Time to restore the BIT to OPERABLE status is consistent with other Completion Times established for loss of a safety function and ensures that the unit will not operate for long periods outside of the safety analyses.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

When Required Action A.1 cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions and to be borated to the required SDM without challenging unit systems or operators. Borating to the required SDM assures that the unit is in a safe condition, without need for any additional boration.

After determining that the BIT is inoperable and the Required Actions of B.1 and B.2 have been completed, the tank must be returned to OPERABLE status within 7 days. These actions ensure that the unit will not be operated with an inoperable BIT for a lengthy period of time. It should be noted, however, that changes to applicable MODES cannot be made until the BIT is restored to OPERABLE status, except as provided by LCO 3.0.4.

C.1

Even though the RCS has been borated to a safe and stable condition as a result of Required Action B.2, either the BIT must be restored to OPERABLE status (Required Action C.1) or the unit must be placed in a condition in which the BIT is not required (MODE 4). The 12 hour Completion Time to reach MODE 4 is reasonable, based on operating experience and normal cooldown rates, and does not challenge unit safety systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

Verification every 24 hours that the BIT water temperature is at or above the specified minimum temperature is frequent enough to identify a temperature change that would approach the acceptable limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. This Frequency has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.6.2

Verification every 7 days that the BIT contained volume is above the required limit is frequent enough to assure that this volume will be available for quick injection into the RCS. The 900 gallon limit corresponds to the BIT being completely full. Methods of verifying that the BIT is completely full include venting from the high point vent, and recirculation flow with the Boric Acid Storage Tanks. If the volume is too low, the BIT would not provide enough borated water to ensure subcriticality during recirculation or to provide additional core shutdown margin following an MSLB. Since the BIT volume is normally stable, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.6.3

Verification every 7 days that the boron concentration of the BIT is within the required band ensures that the reactor remains subcritical following a LOCA; it limits return to power following an MSLB, and maintains the resulting sump pH in an acceptable range so that boron precipitation will not occur in the core. In addition, the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

The BIT is in a recirculation loop that provides continuous circulation of the boric acid solution through the BIT and the boric acid tank (BAT). There are a number of points along the recirculation loop where local samples can be taken. The actual location used to take a sample of the solution is specified in the unit Surveillance procedures. Sampling from the BAT to verify the concentration of the BIT is not recommended, since this sample may not be homogenous and the boron concentration of the two tanks may differ.

The sample should be taken from the BIT or from a point in the flow path of the BIT recirculation loop.

REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.

3.7 PLANT SYSTEMS

3.7.8 Service Water (SW) System

LCO 3.7.8 Two SW System loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW pump inoperable.	A.1 Throttle SW System flow to Component Cooling (CC) heat exchangers.	72 hours
B. Two SW pumps inoperable.	B.1 Throttle SW System flow to CC heat exchangers.	1 hour
	AND	
	B.2 Restore one SW pump to OPERABLE status.	72 hours

distractor (would be true if 2 pps, or if person erroneously assumes need two on your unit [i.e. shared system] or doesn't understand Bases regarding an "operable" loop.

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

See LCO & actions

The SW System is common to Units 1 and 2 and is designed for the simultaneous operation of various subsystems and components of both units. The source of cooling water for the SW System is the Service Water Reservoir. The SW System consists of two loops and components can be aligned to operate on either loop. There are four main SW pumps taking suction on the Service Water Reservoir, supplying various components through the supply headers, and then returning to the Service Water Reservoir through the return headers. Eight spray arrays are available to provide cooling to the service water, as well as two winter bypass lines. The isolation valves on the spray array lines automatically open, and the isolation valves on the winter bypass lines automatically shut, following receipt of a Safety Injection signal. The main SW pumps are powered from the four emergency buses (two from each unit). There are also two auxiliary SW pumps which take suction on North Anna Reservoir and discharge to the supply header. When the auxiliary SW pumps are in service, the return header may be redirected to waste heat treatment facility if desired. However, the auxiliary SW pumps are strictly a backup to the normal arrangement and are not credited in the analysis for a DBA.

During a design basis loss of coolant accident (LOCA) concurrent with a loss of offsite power to both units, one SW loop will provide sufficient cooling to supply post-LOCA loads on one unit and shutdown and cooldown loads on the other unit. During a DBA, the two SW loops are cross-connected at the recirculation spray (RS) heat exchanger supply and return headers of the accident unit. On a Safety Injection (SI) signal on either unit, all four main SW pumps start and the system is aligned for Service Water Reservoir spray operation. On a containment high-high

(continued)

BASES

BACKGROUND (continued)

pressure signal the accident unit's Component Cooling (CC) heat exchangers are isolated from the SW System and its RS heat exchangers are placed into service. All safety-related systems or components requiring cooling during an accident are cooled by the SW System, including the RS heat exchangers, main control room air conditioning condensers, and charging pump lubricating oil and gearbox coolers.

The SW System also provides cooling to the instrument air compressors, which are not safety-related, and the non-accident unit's CC heat exchangers, and serves as a backup water supply to the Auxiliary Feedwater System, the spent fuel pool coolers, and the containment recirculation air cooling coils. The SW System has sufficient redundancy to withstand a single failure, including the failure of an emergency diesel generator on the affected unit.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the SW System is the removal of decay heat from the reactor following a DBA via the RS System.

APPLICABLE SAFETY ANALYSES

The design basis of the SW System is for one SW loop, in conjunction with the RS System, to remove core decay heat following a design basis LOCA as discussed in the UFSAR, Section 6.2.2 (Ref. 2). This prevents the containment sump fluid from increasing in temperature, once the cooler RWST water has reached equilibrium with the fluid in containment, during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid which is supplied to the Reactor Coolant System by the ECCS pumps. The SW System also prevents the buildup of containment pressure from exceeding the containment design pressure by removing heat through the RS System heat exchangers. The SW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SW System, in conjunction with the CC System, also cools the unit from residual heat removal (RHR), as discussed in the UFSAR, Section 5.5.4, (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CC and RHR System trains that are operating.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SW loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW loop is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Either

a.1 Two SW pumps are OPERABLE in an OPERABLE flow path; or

a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled; and

b. Either

b.1 Three spray arrays are OPERABLE in an OPERABLE flow path; or

b.2 Two spray arrays are OPERABLE in an OPERABLE flow path, provided two spray arrays are OPERABLE in the other loop; and the spray valves for the required OPERABLE spray arrays in both loops are secured in the accident position and power removed from the valve operators; and

c. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

A required valve directing flow to a spray array, bypass line, or other component is considered OPERABLE if it is capable of automatically moving to its safety position or if it is administratively placed in its safety position.

FYI, included to support distractor analysis for this incorrect choice. TS Bases discussion makes (1-SW-P-1A) 2nd part incorrect selection plausible since candidate may have knowledge of single failure discussion but may not be clear on it.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SW System are determined by the systems it supports.

ACTIONS

A.1

If one SW System loop is inoperable due to an inoperable SW pump, the flow resistance of the system must be adjusted within 72 hours by throttling component cooling water heat exchanger flows to ensure that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

B.1 and B.2

If one or more SW System loops are inoperable due to only two SW pumps being OPERABLE, the flow resistance of the system must be adjusted within one hour to ensure that design flows to the RS System heat exchangers are achieved if no additional failures occur following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. Two SW pumps aligned to one loop or one SW pump aligned to each loop is capable of performing the safety function if CC heat exchanger flow is properly throttled. However, overall reliability is reduced because a single failure disabling a SW pump could result in loss of the SW System function. The one hour time reflects the need to minimize the time that two pumps are inoperable and CC heat exchanger flow is not properly throttled, but is a reasonable time based on the low probability of a DBA occurring during this time period. Restoring one SW pump to OPERABLE status within 72 hours together with the throttling ensures that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

BASES

ACTIONS (continued)

C.1

If one SW loop is inoperable for reasons other than Condition A, action must be taken to restore the loop to OPERABLE status.

In this Condition, the remaining OPERABLE SW loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW loop could result in loss of SW System function. The inoperable SW loop is required to be restored to OPERABLE status within 72 hours unless the criteria for a 7 day Completion Time are met, as stated in the 72 hour Completion Time Note. The 7 day Completion Time applies if the three criteria in the 7 day Completion Time Note are met.

The first criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if the inoperability of one SW loop is part of SW System upgrades. Service Water System upgrades include modification and maintenance activities associated with the installation of new discharge headers and spray arrays, mechanical and chemical cleaning of SW System piping and valves, pipe repair and replacement, valve repair and replacement, installation of corrosion mitigation measures and inspection of and repairs to buried piping interior coatings and pump or valve house components. The second criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if three SW pumps are OPERABLE from initial Condition entry, including one SW pump being allowed to not have automatic start capability. The third criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if two auxiliary SW pumps are OPERABLE from initial Condition entry. The 72 hour and 7 day Completion Times are both based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this time period. The 7 day Completion Time also credits the redundant capabilities afforded by three OPERABLE SW pumps (one without automatic start capability) and two OPERABLE auxiliary SW pumps. Changing the designation of the three OPERABLE SW pumps during the 7 day Completion Time is allowed.

BASES

ACTIONS (continued)

D.1 and D.2

If the SW pumps or loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

If two SW loops are inoperable for reasons other than only two SW pumps being OPERABLE, the SW System cannot perform the safety function. With two SW loops inoperable, the CC System and, consequently, the Residual Heat Removal (RHR) System have no heat sink and are inoperable. Twelve hours is allowed to enter MODE 4, in which the Steam Generators can be used for decay heat removal to maintain reactor temperature. Twelve hours is reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging unit systems. The unit may then remain in MODE 4 until a method to further cool the units becomes available, but actions to determine a method and cool the unit to a condition outside of the Applicability must be initiated within one hour and continued in a reasonable manner and without delay until the unit is brought to MODE 5.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the SW System flow path provides assurance that the proper flow paths exist for SW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or

(continued)

3. 008AA2.25 078/NEW//H/4/2.8/3.4/3/

Given the following conditions:

- Unit 1 is at 100% power.
- A Reactor Coolant System pressure transient causes a PRZR safety valve to open prematurely.
- The safety valve then sticks in the fully open position.

Which ONE of the following identifies the plant response to this event, **AND** includes the proper prioritization of the procedures listed below?

1-AP-16, Increasing Primary Plant Leakage
1-AP-44, Loss of RCS Pressure
1-E-0, Reactor Trip or Safety Injection
1-ES-0.1, Reactor Trip Response
1-E-1, Loss of Reactor or Secondary Coolant

RCS pressure will decrease requiring a reactor trip, and the rate of inventory loss will ______________.

A. exceed the capacity of maximum normal charging flow - safety injection will be required; the crew should respond with 1-AP-16, then 1-E-0, followed by 1-E-1.

B. exceed the capacity of maximum normal charging flow - safety injection will be required; the crew should respond with 1-AP-44, then 1-E-0, followed by 1-E-1.

C. be within the capacity of maximum normal charging flow - safety injection will NOT be required; the crew should respond with 1-AP-16, then 1-E-0, followed by 1-ES-0.1.

D. be within the capacity of maximum normal charging flow - safety injection will NOT be required; the crew should respond with 1-AP-44, then 1-E-0, followed by 1-ES-0.1.

a. Incorrect. First part is correct (see 'b'). Second part is incorrect since AP-16 assumes loss of inventory will cause PRZR level to decrease; for a vapor space LOCA however PRZR level will increase and AP-16 will NOT provide appropriate mitigating actions. Plausible since AP-16 Entry Conditons are (increasing PRT pressure/temperature) as a consequence of the subject failure; also CNTMT sump pumping rate will increase once the PRT rupture disc blows. The rest of the procedure transition (E-0 to E-1) is correct (see 'b').

b. Correct. First part is correct per UFSAR accident analysis (accidental depressurization of RCS, which states the makeup flow rate from one charging pump is adequate to accomodate a 3/8" diameter break - the inventory loss from the subject stuck open PRZR SV is in excess of this). This is not intuitively obvious however, since even if a person had a relative feel, the 3/8" break is associated with a liquid inventory loss whereas the stuck open PRZR SV represents a steam inventory loss). AP-44 is appropriate since Entry Conditions are met and the procedure will direct a reactor trip due to decreasing RCS pressure; E-0 will then direct manual safety injection and the diagnostic steps will eventually direct transition to E-1 (depending 'a').

d. Incorrect. First part is incorrect as discussed in answer b. Second part is incorrect but plausible since this would be a credible flowpath for smaller amounts of PRZR inventory loss, and also since flow is proportional to D/P pressure could plateau at a value between that of normal operating pressure and the much lower pressure of 1780 psig where SI would be required.

QUESTIONS REPORT

for NAPS 2010 SRO NRC Exam rev3

Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13)

Expected leak rate from open PORV or code safety

Tier: 1
Group: 1

Technical Reference: study guide for TAA, NAPS UFSAR section 15, 1-AP-16, 1-AP-44, EOP E-0

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

OBJ 13961

9.5.1. The BOL case DNBR remains essentially unchanged.

9.5.2. The EOL manual rod control case DNBR experiences a slight reduction since there is a much larger increase in reactor power due to the moderator feedback, but the DNBR remains above the limit value.

9.5.3. Both the BOL and EOL automatic rod control cases experience core power increases, but for both cases the minimum DNBR remains above the limit value.

9.5.4. Therefore, the excessive load increase transient does not present a safety problem for any rod control configuration any time during core life.

10. Accidental depressurization of the Reactor Coolant System.

10.1. The most severe core conditions resulting from the accidental depressurization of the RCS are associated with the inadvertent opening of a PRZR safety valve.

10.2. Potential causes.

10.2.1. Spurious opening (or opening in response to actual RCS overpressure) of a PRZR safety followed by failure to reseat.

10.3. Protection afforded.

10.3.1. PRZR pressure lo-lo pressure reactor trip/safety injection.

10.3.2. Overtemperature delta-T reactor trip.

10.3.3. Tech Spec requirements for operability of RPS, ESF, ECCS, and other safety-related systems.

10.3.4. Abnormal and Emergency Operating Procedures provide guidance to the control room team to mitigate the consequences of the transient.

10.4. Sequence of events.

10.4.1. Assumptions:

10.4.1.1. Beginning of core life

10.4.1.2. Reactor power, RCS pressure and temperature are at nominal full power values.

10.4.1.3. Rod control is in automatic.

10.4.2. Sequence:

10.4.2.1. Rapid decrease in RCS pressure

10.4.2.2. Increase in PRZR level

10.4.2.3. Pressure reaches hot leg saturation pressure

10.4.2.4. Power maintained constant until reactor trips on PRZR low pressure or overtemperature delta-T.

10.5. Affects and consequences.

10.5.1. PRZR low pressure and overtemperature delta-T reactor trips provide adequate protection.

10.5.2. The minimum DNBR remains above the limit.

11. Accidental depressurization of the Main Steam System.

11.1. The most severe core conditions resulting from an accidental depressurization of the MS system are associated with an inadvertent opening of a single steam dump, SG PORV or safety valve.

11.2. Potential causes.

11.2.1. Control system or equipment malfunction resulting in steam dump, PORV, or SV opening and failing to reclose.

11.3. Protection afforded.

11.3.1. SI system actuation due to PRZR low-low pressure or high steamline differential pressure.

11.3.2. Overpower delta-T and high neutron flux reactor trips.

11.3.3. Main feedwater isolation (reactor trip with low Tave, or SI).

11.3.4. Tech Spec requirements for operability of RPS, ESF, ECCS, and other safety-related systems.

11.3.5. Abnormal and Emergency Operating Procedures provide guidance to the control room team to mitigate the consequences of the accident.

11.4. Sequence of events.

11.4.1. Assumptions:

11.4.1.1. End of core life, most reactive rod stuck fully withdrawn, only one charging pump available.

11.4.1.2. Hot shutdown with all steam generators at 1020 psia.

11.4.2. Sequence:

Figures 15.2-47 through 15.2-48 illustrate the transient assuming the reactor is in the automatic control mode. Both the beginning-of-life and the end-of-life cases show that the core power increases. Due to the large increase in core power, the coolant average temperature shows a slight increase for the beginning-of-life case and a small decrease for the end-of-life case. For both cases, the minimum DNBR remains above the limit value.

15.2.11.3 **Conclusions**

It has been demonstrated that for an excessive load increase the minimum DNBR during the transient will not be below the limit value.

15.2.12 Accidental Depressurization of the Reactor Coolant System

15.2.12.1 **Identification of Causes and Accident Description**

The most severe core conditions resulting from an accidental depressurization of the reactor coolant system are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing reactor coolant system pressure until this pressure reaches a value corresponding to the hot-leg saturation pressure. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease, however, throughout the transient. The effect of the pressure decrease would be to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor will be tripped by the following reactor protection system signals:

1. Pressurizer low pressure.
2. Overtemperature delta T.

Long term effects of this type of event, i.e., after reactor trip, are addressed in the analysis of the small break Loss of Coolant Accident (Section 15.3.1).

15.2.12.2 **Analysis of Effects and Consequences**

15.2.12.2.1 Method of Analysis

This analysis is performed to ensure that the reactor protection system provides the required DNBR protection for the most severe reactor coolant system depressurization event. The accidental depressurization transient is analyzed with the detailed digital computer code RETRAN (Reference 9). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The RETRAN computer code calculates pressurizer pressure, core inlet temperature, core inlet flow and power transient. The COBRA code with WRB-1 CHF correlation (Reference 10) is used to calculate the minimum DNBR during the transient.

15.3.1.2 General

A reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated small-break LOCA (SBLOCA) has been performed in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented here, and are in compliance with 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors*. This analysis was performed with the NRC-approved NOTRUMP code (Reference 22) of the Westinghouse LOCA-ECCS evaluation model (Reference 23). The thermal behavior of the fuel was analyzed using the LOCTA-IV code (Reference 24). The analytical techniques used are in full compliance with 10 CFR 50, Appendix K.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factor and the performance of the Emergency Core Cooling System. The details of the small break LOCA analysis are documented in Reference 25.

15.3.1.3 Identification of Causes and Accident Description

A LOCA can result from a rupture of the Reactor Coolant System (RCS) or of any line connected to that system up to the first isolation valve. Ruptures of a small cross section will cause expulsion of the coolant at a rate that can be accommodated by the charging pumps. These pumps maintain an operational water level in the pressurizer, permitting the operator to execute an orderly shutdown.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 3/8-inch diameter hole. This break results in a loss of approximately 21 lbm/sec.

For breaks in the range between 3/8-inch diameter and 1-ft^2 area, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer, resulting in a pressure and fluid level decrease in the pressurizer. Reactor trip occurs when the pressurizer low-pressure trip setpoint is reached. The safety injection system (SIS) is actuated when the pressurizer low-low pressure setpoint is reached, activating the high head safety injection pumps. The SIS actuation and subsequent activation of the Emergency Core Cooling System, which occurs with the SIS signal, assumes the most limiting single failure. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

Dominion

NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-44	LOSS OF REACTOR COOLANT SYSTEM PRESSURE (WITH ONE ATTACHMENT)	19
		PAGE 1 of 5

PURPOSE

To provide operator guidance in the event of a decreasing Pressurizer pressure, but not necessarily a decreasing Pressurizer level

ENTRY CONDITIONS

This procedure is entered when the following conditions exists:

- Annunciator Response “B” Panel F-7, PRZR HIGH-LOW PRESSURE
- Annunciator Response “C” Panel D-1, PRZR SAFETY VALVE OR PORV OPEN
- Reactor Coolant System Pressure less than 2335 psig and PZR PORV open

CONTINUOUS USE

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If RCS pressure is less than 1870 psig any time during this procedure, then 1-E-0, REACTOR TRIP OR SAFETY INJECTION, must be initiated while continuing with this procedure.

* *

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]___	CHECK PRZR PORVs - CLOSED:	☐ Close the PORVs.
	☐ • 1-RC-PCV-1455C (Yes)	☐ <u>IF</u> any PORV cannot be closed, <u>THEN</u> manually close the associated Block Valve.
	☐ • 1-RC-PCV-1456	
		☐ <u>IF</u> any PORV is open <u>AND</u> the associated Block Valve will not close, <u>THEN</u> GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
[2]___	CHECK MASTER PRESSURE CONTROLLER - CONTROLLING PROPERLY (Yes)	☐ Put the controller in MANUAL and adjust as required to stabilize and restore pressure.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3] ___	CHECK PRZR SPRAY VALVES - CLOSED: (Yes)	☐ Manually close valves using controller.
	☐ • 1-RC-PCV-1455A	☐ Verify PRZR spray valves closed.
	☐ • 1-RC-PCV-1455B	IF NOT, THEN place failed valve remote close switch in CLOSE:
		☐ • 1-RC-SOV-1455A, 1-RC-PCV-1455A REMOTE CLOSE SOV
		☐ • 1-RC-SOV-1455B, 1-RC-PCV-1455B REMOTE CLOSE SOV
		IF valves can NOT be closed, THEN do the following:
		☐ a) GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
		☐ b) Stop 1-RC-P-1C.
4. ___	VERIFY ALL PRZR HEATERS - ENERGIZED (Yes)	☐ Manually energize PRZR heaters as required to maintain desired PRZR pressure.
5. ___	CHECK 1-CH-HCV-1311, AUXILIARY SPRAY VALVE - CLOSED (Yes)	

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	CHECK STATUS OF PRZR SAFETY VALVES AND PORVs:	
	☐ a) PRZR Safety Valves - CLOSED (No)	☐ a) Notify SRO of open PRZR Safety Valves.
	☐ b) PRZR PORVs - CLOSED OR ISOLATED (Yes)	☐ b) IF the Block Valve for an open PRZR PORV cannot be closed, THEN send an operator to locally place the NITROGEN and PRZR PWR RV Isolation switches for the affected PORV in DISABLE (Located in Unit 1 Emergency Switchgear Room, behind the Auxiliary Shutdown Panel in Appendix R Cabinet - key required).
*7.___	VERIFY RCS PRESSURE - STABLE OR INCREASING (No)	IF Spray valve is failed open, THEN do the following: ☐ a) IF 1-RC-PCV-1455A failed open, THEN stop 1-RC-P-1A. b) IF 1-RC-PCV-1455B failed open, THEN do the following: ☐ • IF 1-RC-P-1A is running, THEN stop 1-RC-P-1B. ☐ Evaluate for other causes of decreasing pressure. ☐ RETURN TO Step 1.
8.___	VERIFY RCS PRESSURE - NORMAL	☐ Adjust PRZR heaters or spray as required to restore pressure.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.___	REFER TO TECHNICAL SPECIFICATIONS FOR ANY INOPERATIVE EQUIPMENT: ☐ • Tech Spec 3.4.11 ☐ • Tech Spec 3.4.13	
10.___	EVALUATE MALFUNCTION: ☐ a) Determine cause of the malfunction ☐ b) Submit Work Requests	
11.___	RESTORE NON-AFFECTED EQUIPMENT TO NORMAL OPERATION	

- END -

PRESSURIZER SAFETY VALVE OR PORV OPEN

PORV Limit Switches indicate a PORV is open.
Safety Valve Sensors indicate a Safety is open.

1.0 Probable Cause

1.1 RCS high pressure for given plant conditions
1.2 Failure of pressurizer pressure control system
1.3 Improper operation of charging and/or letdown systems
1.4 Improper operation of key switches for NDT over pressure protection system
1.5 Mechanical and/or electrical failure of safety valves and/or PORVS
1.6 Failure in valve monitor system
1.7 Spurious SI while in Mode 4 or 5 with NDT Protection in service
1.8 Failure of 1-CH-FCV-1122 while in Mode 4 or 5 with NDT Protection in service

2.0 Operator Action

ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
2.1	VERIFY RCS PRESSURE - NORMAL (No)	IF pressure high, THEN take action to prevent over pressurization. IF pressure low THEN Go To 1-AP-44.
2.2	VERIFY PORV POSITION INDICATION - NORMAL (Yes)	Determine limit switch malfunction. Submit Work Request, if required.
2.3	VERIFY SAFETY VALVE INDICATION AT THE ACOUSTICAL MONITOR PANEL - NORMAL (No)	Determine reason: * safety valve leaking by * monitor panel malfunction Submit Work Request, if required.

****

CAUTION: IF the Low Temperature Overpressure Protection System (LTOPS) is in service, THEN an uncontrolled RCS mass addition transient must be terminated within ten minutes. PORV N2 Accumulators are sized to allow PORVs to cycle continuously for 10 minutes only.

****

2.4	IF IN MODE 4 OR 5 AND PORV OPENS DUE TO SPURIOUS SI, THEN RESET SI, ISOLATE THE BIT, AND GO TO 1-AP-49, LOSS OF NORMAL CHARGING. NA	IF unable to stop mass addition, THEN stop all RCPs and Charging Pumps and GO TO 1-AP-49, Loss of Normal Charging.

2.5 IF IN MODE 4 OR 5 AND PORV OPENS DUE TO FAILURE OF 1-CH-FCV-1122, THEN ISOLATE CHARGING FLOW BY CLOSING 1-CH-MOV-1289A AND GO TO 1-AP-49, LOSS OF NORMAL CHARGING

N/A

IF unable to stop mass addition, THEN stop all RCPs and Charging Pumps and GO TO 1-AP-49, Loss of Normal Charging.

3.0 References

3.1 Valve monitor tech manual
3.2 Instrument loop manuals for VMS system
3.3 11715-ESK-10AAE (DC-79-54B)
3.4 DCP 84-17

4.0 Actuation

4.1 Y-VMS100A, B, C (Relay K1)
4.2 33 1-RC-PCV-1455C
4.3 33 1-RC-PCV-1456

Dominion

NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-16	INCREASING PRIMARY PLANT LEAKAGE (WITH FIVE ATTACHMENTS)	26
		PAGE 1 of 16

PURPOSE

To provide instructions for locating, quantifying, and mitigating increasing primary plant leakage.

ENTRY CONDITIONS

- This procedure is entered when there is increasing primary plant leakage, as indicated by any of the following:
  - Increasing charging flow or more frequent VCT makeups,
  - Decreasing PRZR level due to leakage, — No! (vapor space LOCA)
  - Increasing PRT pressure, temperature, or level not due to normal operations,
  - Increasing Containment pressure, or temperature, not due to normal operations, ← not initially, but eventually
  - Unexplained increase in RCP Thermal Barrier CC flow or temperature,
  - Increasing Reactor Vessel Flange Leakoff temperature,
  - More frequent PDTT pumping or Containment Sump Pump operation, ← ... eventually
  - Increasing area radiation or radiation in systems interfacing with RCS,
  - Leak Rate PT results increasing, or
  - Unexplained increase in Auxiliary Building sump level.
- As directed by 1-AP-17, SHUTDOWN LOCA

UNIT ONE

CONTINUOUS USE

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: When the source of leakage has been identified and corrected, then continue with Step 18.

1.___	VERIFY UNIT IN MODE 1, 2, OR 3 Yes	Do the following: ☐ a) IF RHR is lost, THEN GO TO 1-AP-11, LOSS OF RHR. ☐ b) IF RCS level is decreasing with the Reactor Head installed, THEN GO TO 1-AP-17, SHUTDOWN LOCA. ☐ c) IF this procedure was entered from 1-AP-17, SHUTDOWN LOCA, THEN GO TO Step 2. ☐ d) IF Refueling Cavity level decreases in an uncontrolled manner, THEN GO TO 1-AP-52, LOSS OF REFUELING CAVITY LEVEL DURING REFUELING.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

*2.___ VERIFY THE FOLLOWING PARAMETERS - UNDER CONTROL OF OPERATOR: → No!

- ☐ • PRZR level - Increasing
- ☐ • RCS subcooling based on Core Exit TCs No!
- ☐ • VCT level

(STEP 2 CONTINUED ON NEXT PAGE)

IF PRZR level is decreasing, THEN do the following:

a) Isolate Letdown by closing the following valves:

1) Letdown Orifice Isolation Valves:

- ☐ • 1-CH-HCV-1200A
- ☐ • 1-CH-HCV-1200B
- ☐ • 1-CH-HCV-1200C

2) Letdown Isolation Valves:

- ☐ • 1-CH-LCV-1460A
- ☐ • 1-CH-LCV-1460B

☐ b) IF PRZR level cannot be maintained in AUTO level control, THEN place 1-CH-FCV-1122 controller in Manual and adjust Charging flow to control PRZR level.

(STEP 2 RNO CONTINUED ON NEXT PAGE)

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	VERIFY THE FOLLOWING PARAMETERS - UNDER CONTROL OF OPERATOR: (Continued)	

~~☐ c) Start a makeup to the VCT from the blender.~~

~~IF PRZR level OR VCT level cannot be maintained with Letdown isolated and one Charging Pump at maximum charging flow, THEN do the following:~~

~~a) Shift Charging Pump Suction to RWST as follows:~~

~~1) Open Charging Pump suction from RWST MOVs:~~

- ~~☐ 1-CH-MOV-1115B~~
- ~~☐ 1-CH-MOV-1115D~~

~~2) Close Charging Pump suction from VCT MOVs:~~

- ~~☐ 1-CH-MOV-1115C~~
- ~~☐ 1-CH-MOV-1115E~~

~~b) IF PRZR level cannot be maintained, THEN do the following, while continuing with this procedure:~~

- ~~☐ IF Unit 1 is in Mode 1, 2, or 3, THEN GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.~~
- ~~☐ IF Unit 1 is in Mode 4, 5, or 6, THEN GO TO 1-AP-17, SHUTDOWN LOCA.~~

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13. ___ **CHECK IF RCS IS INTACT INSIDE CONTAINMENT:**

- ☐ • Containment pressure - NORMAL
- ☐ • Containment Recirc Spray Sump level - NORMAL
- ☐ • Containment radiation - NORMAL

Response Not Obtained: ☐ GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, STEP 1.

14. ___ **CHECK FOR OUTSIDE CONTAINMENT INVENTORY LOSS:**

Response Not Obtained: ☐ Determine cause of abnormal conditions.

☐ a) Vent Stack radiation - NORMAL:

- MGP Vent Stack A

AND

- MGP Vent Stack B

Response Not Obtained: ☐ IF cause is a loss of RCS inventory outside Containment, THEN GO TO 1-ECA-1.2, LOCA OUTSIDE CONTAINMENT, STEP 1.

b) Safeguard Area Sump level annunciators - NOT LIT:

- ☐ • Annunciator Panel "A" C-1
- ☐ • Annunciator Panel "E" F-8

☐ c) Auxiliary Building Sump level - Annunciator Panel "E" F-6 - NOT LIT

☐ d) Ambient Area Temperatures - Annunciator Panel "A" C-4 - NOT LIT

4. 010G2.4.31 079/NEW//H/3/4.2/4.1/3/

Given the following conditions:

- The crew has just completed synchronizing Unit 1 Main Generator to the grid.
- 1-RC-PT-1444, PRZR Pressure Control transmitter, fails HIGH.
- The crew performs the immediate actions of 1-AP-44, Loss of Reactor Coolant System Pressure.

The OATC reports that PRZR spray valve 1-RC-PCV-1455A indicates fully open, even after placing 1-RC-SOV-1455A, 1-RC-PCV-1455A Remote Close SOV, in CLOSE.

Which ONE of the following describes how the crew will mitigate this event?

A. Suspend performance of 1-AP-44, and perform 1-E-0, Reactor Trip or Safety Injection; stop the "A" RCP when directed by 1-E-0.

B. Suspend performance of 1-AP-44, and perform 1-E-0, Reactor Trip or Safety Injection; stop the "C" RCP when directed by 1-E-0.

C. Perform 1-E-0, Reactor Trip or Safety Injection, while continuing with 1-AP-44; after 1-E-0 Immediate Actions are complete, stop the "A" RCP.

D. Perform 1-E-0, Reactor Trip or Safety Injection, while continuing with 1-AP-44; after 1-E-0 Immediate Actions are complete, stop the "C" RCP.

a. Incorrect. Plausible since EOPs normally take precedence and AOPs are performed on a not to interfere with basis; this however is a case where the AP specifically directs concurrent performance. Further, Spray valve status is checked early in E-0 (step 9). Second part is plausible since the candidate who lacks detailed procedure knowledge may rely solely on systems knowledge and erroneously conclude that stopping only the RCP associated with the failed spray valve is adequate.

b. Incorrect. First part incorrect but plausible as noted above; second part is correct regarding the RCPs that would be stopped per E-0, Step 9 RNO, but as noted above this is not the correct procedural flowpath since AP-44 was specifically written to address this condition in order to minimize the likelihood of an undesired automatic safety injection on lo-lo PRZR pressure. Second part was a change recently implemented based on a WOG issue.

c. Incorrect. First part is correct; as previously discussed 1-AP-44 directs the operator to perform E-0 while continuing with this procedure. Second part incorrect but plausible as discussed in Distractor a.

d. Correct. First part is correct as discussed in Distractors b & c; second part is also correct--once the reactor is verified to be successfully tripped the subject RCPs will be stopped per 1-AP-44 to restore RCS pressure.

QUESTIONS REPORT

for NAPS 2010 SRO NRC Exam rev3

Pressurizer Pressure Control System (PZR PCS)

Knowledge of annunciator alarms, indications, or response procedures.
(CFR: 41.10 / 45.3)

Tier: 2
Group: 1

Technical Reference: OP-AP-104, 1-AP-44, & EOP E-0

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

3.7 Abnormal Operating Procedures

> **NOTE:** Except at Millstone, immediate actions are identified clearly as such in Abnormal Operating Procedures.

Procedure User

3.7.1 **PERFORM** immediate actions from memory.

3.7.2 **APPLY** the following communication techniques when using Abnormal Operating Procedures:

- A NOTE or CAUTION is read verbatim when encountered for the first time during an event
- If not applicable, a NOTE or CAUTION need not be verbalized
- If encountered repeatedly, a NOTE or CAUTION may be paraphrased
- A NOTE or CAUTION may be directed to applicable crew members; in such cases, the applicable crew member will paraphrase or acknowledge the NOTE or CAUTION

> **NOTE:** The priority of procedures depends upon the events in progress. Some abnormal operating procedures must be implemented while emergency operating procedures are in effect. In cases of parallel procedure usage, the EOP receives priority and immediate actions are completed before parallel procedure usage. When using an AOP in parallel with the EOP, only those steps in the AOP that ensure success of the EOP are required to be performed.

correct answer parallel performance (with requirement to do EOP IDAs prior to parallel performance)

3.7.3 **REVIEW** the immediate action steps of abnormal operating procedures after performance to ensure required actions have been taken.

3.7.4 **COMPLETE** subsequent action steps of the abnormal operating procedures as listed in the procedures.

3.7.5 __IF__ a variance becomes necessary during performance of abnormal operating procedures, __THEN__ **PERFORM** the following:

a. Obtain concurrence from a second SRO, if possible.

5.2.2 **Unit SRO**

The Unit SRO is responsible for:

a. Directing the actions specified by EOPs and AOPs during transient conditions.

b. Updating the shift team on current plant conditions.

5.2.3 **Operators**

Operators are responsible for performing actions specified by EOPs and AOPs with the primary emphasis on ensuring that all steps are performed correctly, in a deliberate and organized manner, while adhering to the procedure sequence strategy.

5.2.4 **Shift Technical Advisors**

Shift Technical Advisors are responsible for:

a. Monitoring the status of Critical Safety Function Status Trees and reporting status changes to the Unit SRO.

b. Monitoring overall unit and crew response to the accident or transient and reporting any nuclear safety concern to the Shift Manager.

c. Assessing plant conditions, performing an initial assessment of the Safety Functions for entry into any ORP or FRP and report status to the US.

d. Providing the Shift Manager with technical and Emergency Plan assistance and concurrence.

e. Independently monitoring and evaluating Safety Functions.

f. Providing feedback to the US, as necessary, to ensure correct diagnosis.

g. Maintaining an overall event perspective.

5.3 Definitions

5.3.1 **Continuously Applicable Step**

Any step or action that is continuously applicable throughout the procedure or portion of the procedure.

5.3.2 **Critical Safety Function**

An activity which serves to protect the integrity of one or more physical barriers against radiation release.

NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-44	LOSS OF REACTOR COOLANT SYSTEM PRESSURE (WITH ONE ATTACHMENT)	19
		PAGE 1 of 5

PURPOSE

To provide operator guidance in the event of a decreasing Pressurizer pressure, but not necessarily a decreasing Pressurizer level

ENTRY CONDITIONS

This procedure is entered when the following conditions exists:

- Annunciator Response "B" Panel F-7, PRZR HIGH-LOW PRESSURE
- Annunciator Response "C" Panel D-1, PRZR SAFETY VALVE OR PORV OPEN
- Reactor Coolant System Pressure less than 2335 psig and PZR PORV open

UNIT ONE

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If RCS pressure is less than 1870 psig any time during this procedure, then 1-E-0, REACTOR TRIP OR SAFETY INJECTION, must be initiated while continuing with this procedure.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]___	CHECK PRZR PORVs - CLOSED: ☐ • 1-RC-PCV-1455C ☐ • 1-RC-PCV-1456	☐ Close the PORVs. ☐ IF any PORV cannot be closed, THEN manually close the associated Block Valve. ☐ IF any PORV is open AND the associated Block Valve will not close, THEN GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
[2]___	CHECK MASTER PRESSURE CONTROLLER - CONTROLLING PROPERLY	☐ Put the controller in MANUAL and adjust as required to stabilize and restore pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3]___	CHECK PRZR SPRAY VALVES - CLOSED:	☐ Manually close valves using controller.
	☐ • 1-RC-PCV-1455A	☐ Verify PRZR spray valves closed.
	☐ • 1-RC-PCV-1455B	IF NOT, THEN place failed valve remote close switch in CLOSE:
		☐ • 1-RC-SOV-1455A, 1-RC-PCV-1455A REMOTE CLOSE SOV
		☐ • 1-RC-SOV-1455B, 1-RC-PCV-1455B REMOTE CLOSE SOV
		IF valves can NOT be closed, THEN do the following:
	Correct	☐ a) GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
		☐ b) Stop 1-RC-P-1C.
4.___	VERIFY ALL PRZR HEATERS - ENERGIZED	☐ Manually energize PRZR heaters as required to maintain desired PRZR pressure.
5.___	CHECK 1-CH-HCV-1311, AUXILIARY SPRAY VALVE - CLOSED	

See distractor analysis for additional discussion

5. 015A2.03 080/NEW//H/4/3.2/3.5/7/

Unit 1 is at 75% power ramping up to 100% at 0.3% per minute.

AFD is currently indicating zero (0) on all channels.

"CLOSEST"

Which ONE of the following conditions would place the unit closer to an AFD limit, **AND** includes the reason for the Technical Specification limits on AFD?

A. Allowing AFD to drift up to a value of +5 during the ramp;
ensures that the Heat Flux Hot Channel Factor (FQ(Z)) is not exceeded during operation.

B. Allowing AFD to drift down to a value of -5 during the ramp;
ensures that the Heat Flux Hot Channel Factor (FQ(Z)) is not exceeded during operation.

C. Allowing AFD to drift up to a value of +5 during the ramp;
ensures that the gross radial power distribution remains consistent with the design values used in the Safety Analyses.

D. Allowing AFD to drift down to a value of -5 during the ramp;
ensures that the gross radial power distribution remains consistent with the design values used in the Safety Analyses.

a. Correct. First part is correct, per the figure in the COLR this is closer to the limit at full power than the given alternative (-5). Bases is correct, per the applicable safety analysis of TS 3.2.3 bases this is a reason for maintaining the AFD within limits.

b. Incorrect. First part is incorrect but plausible since the candidate who lacks detailed knowledge of the limitations associated with AFD may not be able to discriminate a difference between the given alternatives (-5 vs. +5). Second part is correct as discussed in answer a.

c. Incorrect. First part is correct as discussed in answer a. Second part is incorrect but plausible since this is a function of another TS power distribution limit (QPTR) an the candidate who lacks detailed knowledge of the bases may select this distractor.

d. Incorrect. First part is incorrect but plausible as discussed in Distractor b. Second part is incorrect but plausible as discussed in distractor c.

QUESTIONS REPORT

for NAPS 2010 SRO NRC Exam rev3

Nuclear Instrumentation System (NIS)

Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.5)

Xenon oscillations

Tier: 2
Group: 2

Technical Reference: COLR figure 3.2-2 and TS 3.2.3 bases

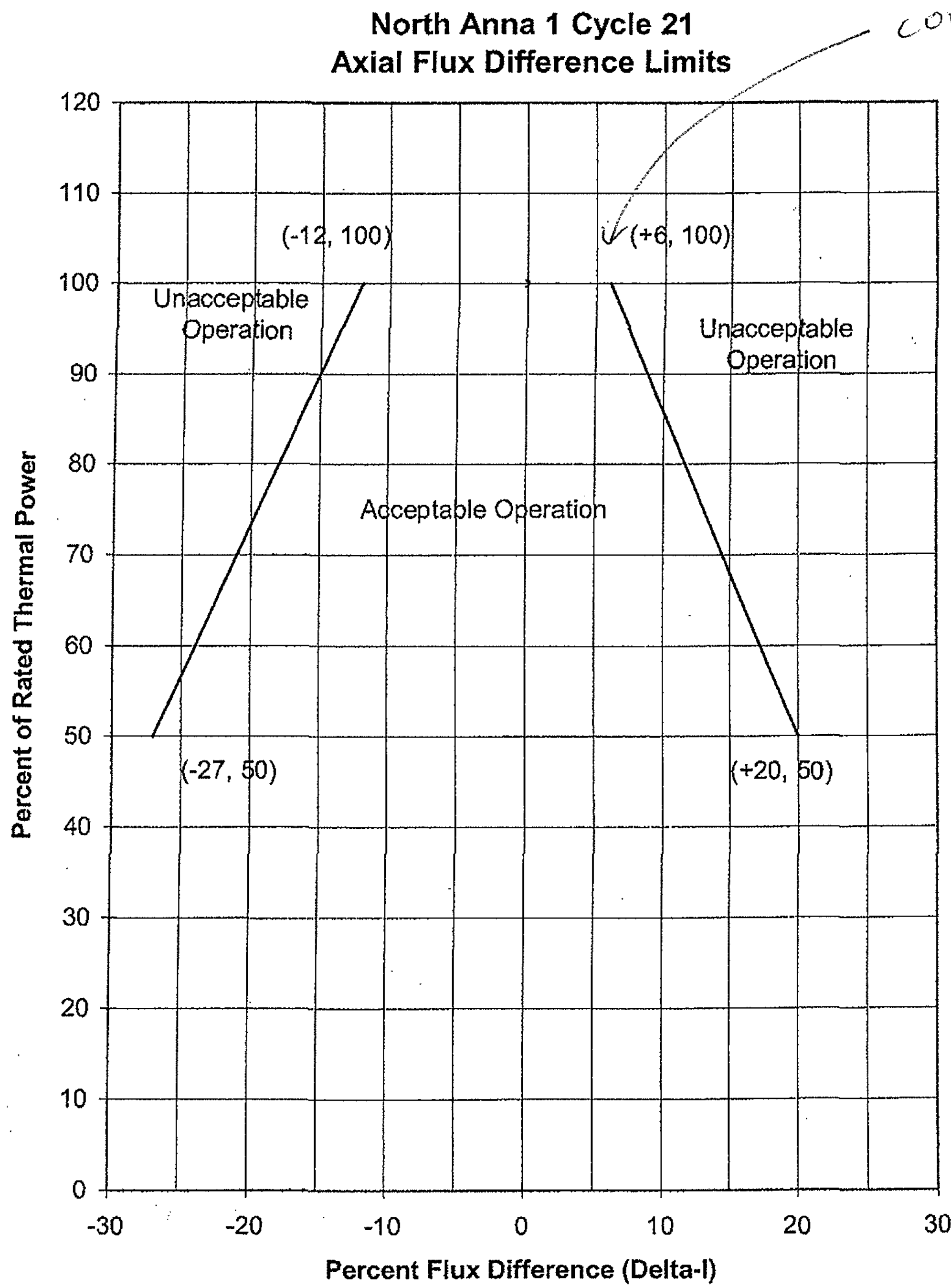
Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: KA match since AFD directly relates to power distribution and thus indirectly to Xenon distribution. AFD is a direct measure of flux distribution, and thus an indirect measure of xenon changes; understanding of the reason for why AFD is maintained within limits demonstrates higher order understanding of need for maintaining core power distributions within limits whether for short term (rod induced transients) or longer term re-distribution induced transients. The facility has no specific procedures for Xenon as the actions for AFD encompass any required actions, activities such as NI cals do not induce xenon oscillations (this was noted since many westinghouse plants in the past did induce oscillations for qtrly cals - that practice is antequated and has been abandoned for the most part).

COLR Figure 3.2-2



correct answer
see distractor analysis for discussion of alternatives.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

See next pg

Relaxed Power Distribution Control (RPDC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RPDC methodology (Ref. 1) establishes a xenon distribution library with tentatively wide AFD limits. Axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

answer

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled rod withdrawal, excessive heat removal, and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 2). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

FYI, supports distractor, see distractor analysis for further discussion

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation by using the movable incore detector system to obtain full core flux maps. Between these full core flux maps, the excore neutron detectors are used to monitor QPTR, which is a measure of changes in the radial power distribution. QPTR is defined in Section 1.1 in terms of ratios of excore detector calibrated output. However, the movable incore detector system can measure changes in the relative power of symmetrically located incore locations or changes in the incore tilt, which can be used to calculate an equivalent QPTR.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature during a small break LOCA must not exceed 2200°F, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

6. 016A2.01 081/NEW//H/3/3.0/3.1/7/

Given the following conditions:

- Unit 1 is at 100% power.
- Containment Pressure Protection Channel I, 1-P-LM-100A, loses power.
- The crew has completed 1-AP-3, Loss of Vital Instrumentation.
- The crew has NOT yet performed any actions of the applicable MOP.

Which ONE of the following identifies the Technical Specification required action, **AND** includes the plant response if Containment Pressure Protection Channel II subsequently fails HIGH prior to any actions being taken?

Place 1-P-LM-100A in BYPASS within ________.

A. 1 hour; an ESF actuation will occur

B. 72 hours; an ESF actuation will occur

C. 1 hour; an ESF actuation will NOT occur

D. 72 hours; an ESF actuation will NOT occur

a. Incorrect. First part is incorrect but plausible since some functions have 1 hr actions associated with them and the candidate who lacks detailed knowledge may select this distractor since it seems more conservative. Second part is incorrect but plausible because most of the ESF functions are de-energize to trip. This channel is associated with the high-high bistables only, which unlike the high and intermediate high-high, functions of the other three containment pressure channels is energize to actuate. So if a different channel were given in the stem, then based on the given failure mode this second part would be correct.

b. Incorrect. First part is correct IAW TS table 3.3.2-1, function 2c, condition E. Second part is incorrect but plausible, as discussed in distractor a.

c. Incorrect. First part is incorrect but plausible as discussed in distractor a; Second part is correct; as previously mentioned the subject channel is unique (energize to actuate function on Hi-Hi only), so based on the given failure mode (loss of power) a susequent high failure on an additional channel will not cause actuation.

d. Correct. First part is correct as discussed in Distractor b. Second part is also correct as discussed in distractor c.

Non-Nuclear Instrumentation System (NNIS)

Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.5)

Detector failure

Tier: 2
Group: 2

Technical Reference: dwg 1082H41 sh. 19 of 29 & TS table 3.3.2-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: NOTE: 1-AP-3 directs the operator to perform 1-MOP-55.75 within 72 hours so this is the procedure that corrects/addresses the given conditions; since the subject MOP implements the requirements of Tech Specs, "technical specification required action" is used in the stem for simplicity/clarity and readability.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

------------------------------------ NOTE ------------------------------------

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Be in MODE 5.	84 hours

See pg 3.3.2-9

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	C.1 --------NOTE--------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------------------	
	Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 Be in MODE 5.	60 hours
D. One channel inoperable.	D.1 --------NOTE--------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ---------------------	
	Place channel in trip.	72 hours
	OR	
	D.2.1 Be in MODE 3.	78 hours
	AND	
	D.2.2 Be in MODE 4.	84 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One Containment Pressure channel inoperable.	E.1 ---------NOTE--------- One additional channel may be bypassed for up to 12 hours for surveillance testing. ---------------------	
	Place channel in bypass.	72 hours
	OR	
	E.2.1 Be in MODE 3.	78 hours
	AND	
	E.2.2 Be in MODE 4.	84 hours
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	F.2.1 Be in MODE 3.	54 hours
	AND	
	F.2.2 Be in MODE 4.	60 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. One train inoperable.	G.1	--------NOTE--------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------------------	
		Restore train to OPERABLE status.	24 hours
	OR		
	G.2.1	Be in MODE 3.	30 hours
	AND		
	G.2.2	Be in MODE 4.	36 hours
H. One Main Feedwater Pumps trip channel inoperable.	H.1	Restore channel to OPERABLE status.	48 hours
	OR		
	H.2	Be in MODE 3.	54 hours
I. One channel inoperable.	I.1	--------NOTE--------- One additional channel may be bypassed for up to 12 hours for surveillance testing. ---------------------	
		Place channel in bypass.	72 hours
	OR		
	I.2.1	Be in MODE 3.	78 hours
	AND		
	I.2.2	Be in MODE 5.	108 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. One or more channels inoperable.	J.1 Verify interlock is in required state for existing unit condition.	1 hour
	<u>OR</u>	
	J.2.1 Be in MODE 3.	7 hours
	<u>AND</u>	
	J.2.2 Be in MODE 4.	13 hours

plausible 1 hr action but not for this channel
*

SURVEILLANCE REQUIREMENTS

- - - - - - - - - - NOTE - - - - - - - - - -
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.
- -

SURVEILLANCE	FREQUENCY
SR 3.3.2.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.3 Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.4 Perform COT	92 days

* also if person doesn't process series of events correctly they may conclude loss of power causes a channel trip and 2nd instrument failure would cause inst atws and T.S. 3.0.3 would apply with 1hr to act.

SURVEILLANCE		FREQUENCY
SR 3.3.2.5	-------------------NOTE-------------------- Not required to be performed for SLAVE RELAYS if testing would: 1. Result in an inadvertent Reactor Trip System or ESFAS Actuation if accompanied by a single failure in the Safeguard Test Cabinet; 2. Adversely affect two or more components in one or more ESFAS system(s); or 3. Create a reactivity, thermal, or hydraulic transient condition in the Reactor Coolant System. --- Perform SLAVE RELAY TEST.	92 days
SR 3.3.2.6	-------------------NOTE-------------------- Verification of relay setpoints not required. --- Perform TADOT.	92 days
SR 3.3.2.7	-------------------NOTE-------------------- Verification of setpoint not required for manual initiation or interlock functions. --- Perform TADOT.	18 months
SR 3.3.2.8	-------------------NOTE-------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	18 months

SURVEILLANCE		FREQUENCY
SR 3.3.2.9	--------------------NOTE-------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is ≥ 1005 psig. --	
	Verify ESFAS RESPONSE TIMES are within limit.	18 months on a STAGGERED TEST BASIS

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.7	NA
b. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure—High	1, 2, 3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 17.7 psia
d. Pressurizer Pressure—Low—Low	1, 2, 3[a]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 1770 psig
e. High Differential Pressure Between Steam Lines	1, 2, 3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 112 psid
f. High Steam Flow in Two Steam Lines	1, 2, 3[b]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	(c)
Coincident with T_{avg}—Low Low	1, 2, 3[b]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 542°F
g. High Steam Flow in Two Steam Lines	1, 2, 3[b]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	(c)
Coincident with Steam Line Pressure—Low	1, 2, 3[b]	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 585 psig

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Above the P-12 (T_{avg}—Low Low) interlock.

(c) Less than or equal to a function defined as ΔP corresponding to 42% full steam flow below 20% load, and ΔP increasing linearly from 42% full steam flow at 20% load to 111% full steam flow at 100% load, and ΔP corresponding to 111% full steam flow above 100% load.

Table 3.3.2-1 (page 2 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Containment Spray Systems					
a. Manual Initiation	1, 2, 3, 4	2 per train, 2 trains	B	SR 3.3.2.7	NA
b. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure					
High High	1, 2, 3	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 28.45 psia
See pg 3.3.2-5 for distraction					
d. Refueling Water Storage Tank (RWST) Level–Low	1, 2, 3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 59% and ≤ 61%
Coincident with Containment Pressure–High High	Refer to Function 2.c (Containment Spray-Containment Pressure–High High) for all functions and requirements.				
3. Containment Isolation					
a. Phase A Isolation					
(1) Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.7	NA
(2) Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b. Phase B Isolation					
(1) Manual Initiation	Refer to Function 2.a (Containment Spray–Manual Initiation) for all functions and requirements.				
(2) Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
(3) Containment Pressure					
High High	Refer to Function 2.c (Containment Spray–Containment Pressure High High) for all functions and requirements.				

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	1, 2(d), 3(d)	2 per steam line	F	SR 3.3.2.7	NA
b. Automatic Actuation Logic and Actuation Relays	1, 2(d), 3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure–Intermediate High High	1, 2(d), 3(d)	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 18.5 psia
d. High Steam Flow in Two Steam Lines	1, 2(d), 3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	(c)
Coincident with T_{avg}–Low Low	1, 2(d), 3(b)(d)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 542°F
e. High Steam Flow in Two Steam Lines	1, 2(d), 3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	(c)
Coincident with Steam Line Pressure–Low	1, 2,(d) 3(d)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 585 psig
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	1, 2(e), 3(e)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
b. SG Water Level–High High (P-14)	1, 2(e), 3(e)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 76%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(b) Above the P-12 (T_{avg}–Low Low) interlock.

(c) Less than or equal to a function defined as ΔP corresponding to 42% full steam flow below 20% load, and ΔP increasing linearly from 42% full steam flow at 20% load to 111% full steam flow at 100% load, and ΔP corresponding to 111% full steam flow above 100% load.

(d) Except when all MSTVs are closed and de-activated.

(e) Except when all Main Feedwater Pump Discharge Valves or all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	1, 2, 3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
b. SG Water Level–Low Low	1, 2, 3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 17%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
d. Loss of Offsite Power	1, 2, 3	1 per bus, 2 buses	F	SR 3.3.2.6 SR 3.3.2.8 SR 3.3.2.9	≥ 2184 V
e. Trip of all Main Feedwater Pumps	1, 2	2 per pump	H	SR 3.3.2.7 SR 3.3.2.9	NA
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
b. RWST Level–Low Low	1, 2, 3, 4	4	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 15% and ≤ 17%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1, 2, 3	1 per train, 2 trains	F	SR 3.3.2.7	NA
b. Pressurizer Pressure, P-11	1, 2, 3	3	J	SR 3.3.2.1 SR 3.3.2.8	≤ 2010 psig
c. T_{avg}–Low Low, P-12	1, 2, 3	1 per loop	J	SR 3.3.2.1 SR 3.3.2.8	≥ 542°F and ≤ 545°F

Init **Verif**

5.0 INSTRUCTIONS

5.1 Placing Containment Pressure Protection Channel I (P-LM-100A) in Test

______ 5.1.1 Verify Initial Conditions are satisfied.

______ 5.1.2 Review Precautions and Limitations.

CAUTION

IF one of the coincident channel bypass annunciators is LIT, THEN the unit must be placed in the mode required by Tech Spec 3.0.3, due to having less than minimum channels operable for Containment Spray, Containment Phase "B" isolation.

NOTE: This channel does NOT provide input to Safety Injection OR Steam Line Isolation.

NOTE: Because placing this channel in TEST will put the channel in the TEST BYPASS mode and block any output to the Reactor Protection System, no coincidence requirements exist for this channel.

5.1.3 Verify coincident channels are not in bypass by verifying the following annunciators are NOT LIT:

______ • Panel "K" H-4, CONTAINMENT DEPRESSURIZTN ACT BISTABLE BYPASSED

______ • Panel "N" D-6, CNTMT PRESS HI-HI TEST BYP CHNL II

______ • Panel "N" D-7, CNTMT PRESS HI-HI TEST BYP CHNL III

______ • Panel "N" D-8, CNTMT PRESS HI-HI TEST BYP CHNL IV

______ 5.1.4 Using Attachment 1, Instrument Rack Room Cabinet Layout, go to the Instrument Rack Room and locate Channel I Protection Cabinet A.

______ 5.1.5 Unlock and open Channel I Protection Cabinet A and verify that annunciator Panel "P" G-5, PCC CAB I VIOLATED DOOR OPEN, is LIT.

NOTE: Attachment 2, Balance of Plant Typical 2-Bay Cabinet, will aid in identifying the correct card and Attachment 3, Process Typical Channel Test Card, will aid in identifying the correct switch on the card.

NOTE: The Channel Test Status light on the top edge of the Channel Test Card comes on only when the loop is properly in TEST with the master test switch in NORMAL. Attachment 3 will aid in identifying the Channel Test Status light.

5.1.6 Place the following Bistable (BS) switch in TEST and verify the associated annunciators are LIT:

______ ______ SV • CA-424, BS-3 (1-LM-PTS-100A-3)

______ •• Panel "K" H-4, CONTAINMENT DEPRESSURIZTN ACT BISTABLE BYPASSED

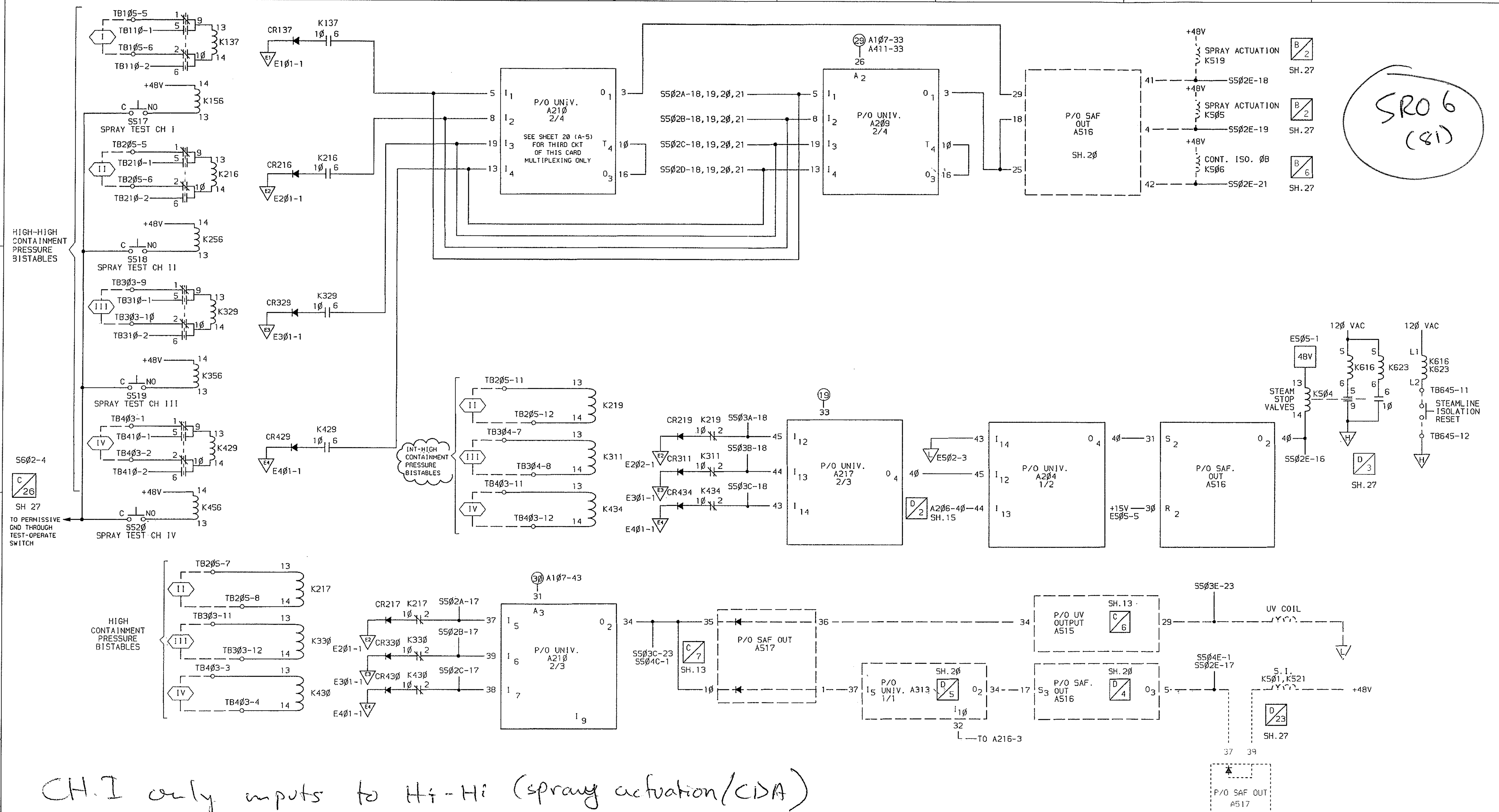
______ •• Panel "N" D-5, CNTMT PRESS HI-HI TEST BYP CHNL I

______ 5.1.7 Close and lock Channel I Protection Cabinet A and verify that annunciator Panel "P" G-5, PCC CAB I VIOLATED DOOR OPEN, is NOT LIT.

______ 5.1.8 Record the failed instrument channel in the Action Statement Status Log.

______ 5.1.9 Notify the Instrument Department that Containment Pressure Protection Channel I (P-LM-100A) failed and has been placed in trip.

______ Completed by: ______________________ Date: __________



SRO 6 (81)

CH. I only inputs to Hi-Hi (spray actuation/CDA)
As shown logic is 2/4, energize to actuate

REVISION DESCRIPTION	DSGN
REVISED PER DCR 2009-1038 THIS DWG SUPERSEDES REV 4	AEG

CAD NO: olon001 H41019.HYB

REF DWG NA-DW-1082H41 SH.1

PC=ND1

SOLID STATE PROTECTION SYSTEM

CONTAINMENT PRESSURE
UNITS 1 & 2

Dominion

NUCLEAR ENGINEERING

NORTH ANNA POWER STATION

DRAWING NUMBER NA-DW-1Ø82H41 SH 19 OF 29 REVISION 5

PRIOR TO USING FOR DESIGN WORK CHECK DMIS FOR WORK PENDING

7. 025AG2.2.36 082/NEW//H/3/3.1/4.2/4/

Given the following conditions:

- Unit 1 core on-load is in progress.
- Reactor cavity water level is 289' 10".
- "A" RHR pump is in operation.
- "B" RHR pump is tagged for seal repairs.

Electrical Maintenance requests to perform a work order to locally observe operation of breaker 15H14 for "A" RHR pump. The work requires the breaker to be opened, and then re-closed, using the control room switch. No test equipment will be attached to the breaker, and the entire operation should take no more than 30 minutes.

Which ONE of the following identifies how the SRO should respond to the Electrical Maintenance request, **AND** includes the Bases for this response?

A. Allow the activity. Tech Specs permit this since a duration time of 2 hours for an inoperable pump has been previously evaluated and determined to be safe; therefore the activity is acceptable per TS 3.9.5.

B. Allow the activity. Tech Specs provide an allowance which permits removing the required operating RHR loop from operation for up to 1 hour per 12 hour period to accomodate maintenance on the RHR loop; therefore the activity is acceptable per TS 3.9.5.

C. DO NOT allow the activity. The bases for allowing the required RHR loop to be removed from service does NOT apply since cavity water level is not sufficient to remove decay heat during the proposed time period; therefore the activity is NOT acceptable per TS 3.9.5,

D. DO NOT allow the activity. The bases for allowing the required RHR loop to be removed from service does NOT apply since it is to permit operations (e.g. fuel movement, core mapping, etc.) in the vicinity of the Hot Leg nozzles; therefore the activity is NOT acceptable per TS 3.9.5.

a. Incorrect. Plausible since the provision does exist. However detailed knowledge of the provisions of TS is required to discern the circumstances for which this allowance applies (for ts 3.9.6 there is a 2 hour provision for survaillance testing), otherwise the candidate may select this distactor.

b. Incorrect. Plausible since there is a provision in the TS but again detailed knowledge of the bases is required since the provision is specific to refueling operations and RHR loop isolation valves; thus the candidate who lacks detailed knowledge of the bases may erroneously conclude that the activity is permissible.

c. Incorrect. Plausible since the candidate who lacks detailed knowledge of the requirements for the given plant conditions may default to not allowing it but may not be clear on the requirements of TS as it relates to the specific bases and rationalize that this would be a logical justification.

d. Correct. As discussed in distractor b, there is an allowance for shutting down RHR, however the bases is very specific as to what activities the allowance applies to and the given condition is not one of them. The action when RHR loop requirements are not met requires initiating action to restore the RHR loop to operation immediately, thus precluding voluntary entry in to the TS action for the given maintenance activity.

Loss of Residual Heat Removal System (RHRS)

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
(CFR: 41.10 / 43.2 / 45.13)

Tier: 1
Group: 1

Technical Reference: TS 3.9.5 & Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: vision #12006

Question History:

additional info:

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

---------- NOTE ----------

The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System (RCS), coolant of boron concentration less than required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close equipment hatch and secure with four bolts.	4 hours
	<u>AND</u>	
	A.5 Close one door in each installed air lock.	4 hours
	<u>AND</u>	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	<u>OR</u>	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.1 Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	12 hours

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit removal of the RHR loop from operation for short durations, under the condition that the boron concentration is not diluted. This conditional removal from operation of the RHR loop does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be

(continued)

BASES

LCO (continued)

OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the RHR discharge temperature. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

BASES

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If LCO 3.9.5 is not met, the following actions must be taken:

a. the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;

b. one door in each installed air lock must be closed; and

(continued)

BASES

ACTIONS

A.4, A.5, A.6.1, and A.6.2 (continued)

c. each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. UFSAR, Section 5.5.4.

8. 026A2.02 083/NEW//H/3/4.2/4.4/5/

Unit 1 experiences a large-break LOCA. The crew was unable to align a flowpath from the containment sump, and has transitioned to 1-ECA-1.1, Loss of Emergency Coolant Recirculation.

The following conditions exist:

- RWST level is 12% and slowly decreasing.
- Containment sump level is 5 foot 10 inches.

Given these conditions, 1-ECA-1.1 will direct the crew to start ____________, and cautions the crew that Quench Spray pumps must be stopped when RWST level decreases to ____________.

A. ONLY the Outside Recirc Spray pumps ; 3%

B. ONLY the Outside Recirc Spray pumps ; 8%

C. ALL available Recirc Spray pumps ; 3%

D. ALL available Recirc Spray pumps ; 8%

a. Incorrect. First part incorrect but plausible since for the typical DBA this would be adequate for containment pressure control at this stage of the event, in this case however all are operated if available to maximize heat removal. Further, one of the functions of the quench spray pumps (which will be subsequently stopped by this procedure) is to provide ~ 150 gpm to the suction of the inside recirc spray pumps for NPSH, so the candidate who relys on systems knowledge may default to this distractor based on that knowledge also. Second part is correct

b. Incorrect. First part incorrect but plausible as noted above. Second part is incorrect, but plausible since this is the level at which charging and low-head pumps are required to be stopped.

c. Correct. First part is correct as discussed in distractor a; ECA-1.1 directs the operator to start all available recirc spray pumps. Second part is correct, pumps can be run down to this level and then shutoff since they will not have adequate NPSH and may experience damage if run for an excessive period in that condition.

d. Correct. First part is correct as discussed in answer c. Second part is incorrect but plausible as discussed in distractor b.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Containment Spray System (CSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Failure of automatic recirculation transfer

Tier: 1
Group: 1

Technical Reference: EOP ECA-1.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

Dominion

NORTH ANNA POWER STATION

EMERGENCY CONTINGENCY ACTION

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-1.1	LOSS OF EMERGENCY COOLANT RECIRCULATION (WITH TWO ATTACHMENTS)	16 PAGE 1 of 34

PURPOSE

To provide instructions to attempt to restore emergency coolant recirculation capability, to delay RWST depletion by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow.

ENTRY CONDITIONS

This procedure is entered from:

- 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT,
- 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, or
- 1-ECA-1.2, LOCA OUTSIDE CONTAINMENT.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- If emergency coolant recirculation capability is restored to at least one train during this procedure, then further recovery actions should continue by returning to the procedure and step in effect.
- If the suction source is lost to any SI Pump or Recirc Spray Pump, then the pump should be stopped.

* *

NOTE:
- IF Containment Sump Blockage has occurred, THEN FRs should NOT be implemented until directed in this procedure.
- ATTACHMENT 2, MINIMUM SI FLOW RATE VERSUS TIME AFTER TRIP, provides adequate injection flow required.
- Setpoints in brackets [ ] are for adverse Containment atmosphere (20 psia Containment pressure or Containment Radiation has reached or exceeded 1.0E5 R/hr or 70% on High Range Recorder).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK EMERGENCY COOLANT RECIRCULATION EQUIPMENT - AVAILABLE: ☐ • Low-Head SI Pumps ☐ • Low-Head SI Pump Suction Valves from Containment	Try to restore at least one train of Emergency Coolant Recirculation Equipment: ☐ • Local operations ☐ • Electrical restoration ☐ • Equipment repair
2.___	RESET BOTH TRAINS OF SI IF NECESSARY	☐ Perform 1-AP-0, RESETTING SI LOCALLY, while continuing with this procedure.
3.___	PUSH BOTH SI RECIRC MODE RESET BUTTONS	
* 4.___	CHECK RWST LEVEL - GREATER THAN 8%	☐ Implement FRs as applicable. ☐ GO TO Step 30.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**ESTABLISH ONE TRAIN OF SI FLOW:**	
	☐ a) Charging Pumps - ONLY ONE RUNNING	☐ a) IF both Low-Head SI Pumps are stopped due to Containment Sump blockage OR loss of suction, THEN GO TO Step 6. ☐ Start or stop Charging Pumps to establish only one pump running.
	☐ b) Place all non-running Charging Pumps in PTL	
	☐ c) RCS pressure - LESS THAN 225 psig [450 psig]	☐ c) Stop both Low-Head SI Pumps. ☐ GO TO Step 7.
	☐ d) Low-Head SI Pumps - ONLY ONE RUNNING	☐ d) IF both Low-Head SI Pumps are stopped due to Containment Sump blockage OR loss of suction, THEN GO TO Step 6. Start or Stop Low-Head SI Pumps, as follows: ☐ • IF Recirc Spray sump level is greater than 8 ft 0 in, THEN start Low-Head SI Pumps to establish only one pump running. OR ☐ • Stop Low-Head SI Pumps to establish only one pump running.
	☐ e) GO TO Step 7	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**ALIGN CHARGING PUMP SUCTION TO RWST**	
	☐ a) Check RWST level - GREATER THAN 8%	☐ a) Implement FRs as applicable. ☐ GO TO Step 30.
	b) Close LHSI Discharge to Charging Pumps: ☐ • 1-SI-MOV-1863A ☐ • 1-SI-MOV-1863B	☐ b) Locally close valves.
	c) Open Charging Pump Suction from RWST Isolation Valves: ☐ • 1-CH-MOV-1115B ☐ • 1-CH-MOV-1115D	☐ c) Locally open at least one valve.
	☐ d) Start one Charging Pump	☐ d) Implement FRs as applicable. ☐ GO TO Step 7.
	☐ e) Verify minimum SI flow required using ATTACHMENT 2, MINIMUM SI FLOW RATE VERSUS TIME AFTER TRIP	☐ e) IF SI flow is less than minimum SI flow, THEN start one additional Charging Pump.
	f) Open Chemical Addition Tank Outlet Valves: ☐ • 1-QS-MOV-102A ☐ • 1-QS-MOV-102B	☐ f) Locally open valves.
	☐ g) Implement FRs as applicable	

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**CHECK RECIRC SPRAY SYSTEM:**	
	☐ a) Recirc Spray Sump level - GREATER THAN 4 FT 10 IN	☐ a) WHEN Recirc Spray Sump is greater than 4 ft 10 in, THEN do Step 7b. ☐ Continue with Step 8.
	answer	
	☐ b) Start all available Recirc Spray Pumps	
	c) Verify SW Supply to RSHX Isolation Valves for running Recirc Spray Pumps - OPEN:	☐ c) Manually open valves.
	☐ • 1-SW-MOV-103A (1-RS-P-1A)	
	☐ • 1-SW-MOV-103D (1-RS-P-2A)	
	☐ • 1-SW-MOV-103B (1-RS-P-1B)	
	☐ • 1-SW-MOV-103C (1-RS-P-2B)	

STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

distractor

* *

CAUTION:
- Charging and Low-Head Pumps taking suction from the RWST must be stopped when RWST level decreases to 8%.
- Quench Spray Pumps taking suction from the RWST must be stopped when RWST level decreases to 3%.

* *

answer

8.___ **DETERMINE CONTAINMENT SPRAY REQUIREMENTS:**

☐ a) Determine number of Quench Spray Pumps required from table:

RWST LEVEL	CONTAINMENT PRESSURE	RECIRC SPRAY PUMPS RUNNING	QUENCH SPRAY PUMPS REQUIRED
Greater than 16%	Greater than 60 psia	N/A	2
	Between 13 psia and 60 psia	Less than 2	2
		2 or more	0
	Less than 13 psia	N/A	0
Between 3% and 16%	Greater than 60 psia	N/A	2
	Between 28 psia and 60 psia	Less than 2	2
		2 or more	0
	Less than 28 psia	ANY	0

(STEP 8 CONTINUED ON NEXT PAGE)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Quench Spray (QS) System

BASES

BACKGROUND

The QS System is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure within 6 hours following a Design Basis Accident (DBA). Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, a dedicated spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Features (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the QS System.

The QS System is actuated either automatically by a containment High-High pressure signal or manually. The QS System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Each train of the QS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The QS System also provides flow to the Inside RS pumps to improve the net positive suction head available.

Supports that distractor only outside recirc spray pumps would be started →

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

(continued)

BASES

BACKGROUND (continued)

cooling tank. The casing cooling pumps are considered part of the outside RS subsystems. Each casing cooling pump is powered from a separate ESF bus.

The inside RS subsystem pump NPSH is increased by reducing the temperature of the water at the pump suction. Flow is diverted from the QS system to the suction of the inside RS pump on the same safety train as the quench spray pump supplying the water.

Supports distractor that only the outside pumps would be started since QS pps will be stopped

The RS System provides a spray of subcooled water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Upon receipt of a High-High containment pressure signal, the two casing cooling pumps start, the casing cooling discharge valves open, and the RS pump suction and discharge valves receive an open signal to assure the valves are open. Refueling water storage tank (RWST) Level—Low coincident with Containment Pressure—High High provides the automatic start signal for the inside RS and outside RS pumps. Once the coincidence logic is satisfied, the outside RS pumps start immediately and the inside RS pumps start after a 120-second delay. The delay time is sufficient to avoid simultaneous starting of the RS pumps on the same emergency diesel generator. The coincident trip ensures that adequate water inventory is present in the containment sump to meet the RS sump strainer functional requirements following a loss of coolant accident (LOCA). The RS system is not required for steam line break (SLB) mitigation. The RS pumps take suction from the containment sump and discharge through their respective spray coolers to the spray headers and into the containment atmosphere. Heat is transferred from the containment sump water to service water in the spray coolers.

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution to the RWST water supplied to the suction of the QS System pumps. The NaOH added to the QS System spray ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the RS spray (pumped from the sump) enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

(continued)

9. 027G2.4.30 084/NEW//H/4/2.7/4.1/5/

Given the following conditions:

- A cavity seal ring failure causes a loss of reactor cavity water level during core off-load.
- Fuel failures during the previous cycle result in high airborne iodine concentration.
- Water level in the Containment basement stabilizes at four feet deep after the cavity drains.
- Painting was recently completed inside Containment.

Which ONE of the following identifies the appropriate actions to reduce containment airborne iodine concentration, **AND** includes the Technical Specification implications?

A. Place Containment Purge in service through one Auxiliary Building Iodine Filter; enter TS 3.7.15 action for Fuel Building Ventilation System inoperable.

B. Place Iodine Filtration Fans 1-HV-F-3A and 3B in service; enter TS 3.7.15 action for Fuel Building Ventilation System inoperable.

C. Place Containment Purge in service through one Auxiliary Building Iodine Filter; enter TS 3.7.12 action for one PREACS train inoperable.

D. Place Iodine Filtration Fans 1-HV-F-3A and 3B in service; enter TS 3.7.12 action for one PREACS train inoperable.

a. Incorrect. First parts is correct, but TS 3.7.15 action does not apply. Plausible since the Fuel Building Ventilation System includes the auxiliary building iodine filters.

b. Incorrect. First part is incorrect but plausible since these fans could be used for cleanup under normal operating conditions; thus the candidate who lacks detailed knowledge of the mitigation strategy based on the given conditions may default to this distractor, especially given that painting has taken place. Second part is incorrect but plausible as discussed above.

c. Correct. Per 0-OP-21.5, Step 5.1.4, iodine filters may be used to mitigate a radiological event if paint fumes are present, but TS 3.7.12 action must be entered. However the candidate must have detailed knowledge of the procedure to possitively conclude this since ultimately there are several factors (such as concentration) which are ultimately used to determine whether the filters continue to perform there design bases function for their required accident mission time.

d. Incorrect. First part is incorrect but plausible as discussed in distractor b. Second part is correct, for the correct lineup, but plausible for this situation if the candidate applies the same conservative logic (i.e. based on potential for filter contamination from paint fumes) that the train should be declared inoperable.

Containment Iodine Removal System (CIRS)

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.
(CFR: 41.10 / 43.5 / 45.11)

Tier: 1
Group: 1

Technical Reference: 0-OP-21.5, 1-OP-21.2, TS 5.5.10, TS 3.7.12, ET SE-99-046

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History: New

additional info: NAPS containment iodine removal units are located in the containment basement (below the flood level) and are not post-accident (or TS) equipment. Their function and design is for pre-outage atmospheric cleanup. Reportability portion of K/A is met since entry into TS action is reportable internally.

2.3.6 ET SE-99-046, Rev. 0, Terms used in Tech Spec Sections 4.6.4.3.b, 4.7.7.1.b, and 4.7.8.1.b Regarding Charcoal Filters

2.3.7 Safety Evaluation 96 SE-OT-28, included with Rev. 18

2.4 **Commitment Documents**

2.4.1 CTS Assignment 02-91-2805-003, SER 1-91, Spent Fuel Pool Overflow Events

2.4.2 CTS Assignment 02-94-2220-001, Surry Enforcement Conference - 9/30/94, Charcoal Filter Degradation

2.4.3 PI N-2003-1252-R1, Mis-Communications Between Ops and HP

2.4.4 Plant Issue N-2005-4706-R1, HP not notified of release of Gas Stripper

2.4.5 LER N1-96-002-00, Containment Radiation Monitors Inoperable Due to Air Recirculating Fans Not Operating

Init Verif

3.0 INITIAL CONDITIONS

_______ 3.1 Review the equipment status to verify station configuration supports the performance of this procedure.

_______ 3.2 A known atmospheric condition exists in the containment as sampled by Health Physics.

_______ 3.3 Reactor is shutdown, in either mode 5 or 6 or defueled.

_______ 3.4 IF in Mode 6, THEN at least one Containment Air Recirc Fan is in operation to provide representative sample for 1-RM-RMS-159 and 1-RM-RMS-160. **(Reference** 2.4.5)

_______ 4.8 Better containment cooling with a containment access hatch open or removed can be obtained by operating Containment Purge System with two Exhaust Fans running and no Supply fans running. This allows the relatively cool outside air to flow in through the hatch and flow down to the exhaust fan suctions.

_______ 4.9 Due to possible Charcoal Filter degradation, the Auxiliary Building Iodine Filters MUST NOT be placed in service during or following painting, fire, or chemical release in any ventilation zone communicating with the inlet to any Auxiliary Building Iodine Filters which will be operated. IF the Auxiliary Building Iodine Filters MUST be placed in service under these conditions, such as to mitigate radiological events, THEN the surveillance requirements of Tech Spec 5.5.10 and Ventilation Filter Testing Program (VFTP) must be satisfied. **(References 2.4.2 and 2.3.6)**

_______ 4.10 At least one Containment Air Recirc Fan must be in operation to provide a representative sample for 1-RM-RMS-159 and 1-RM-RMS-160. **(Reference 2.4.5)**

_______ 4.11 Sample flow for 1-RM-RMS-159 and 1-RM-RMS-160 must be between 8 cfm and 12 cfm to be considered operable to satisfy Technical Requirements Manual TR 3.3.7. **(References 2.3.4. and 2.3.7)**

NOTE: When it is desired to establish atmospheric conditions in the containment, do NOT under any circumstances open 1-HV-MOV-100B, CONT PURGE SUPPLY VALVE (outside valve), while vacuum is being broken. This to prevent the possible collapse of purge system duct work.

5.1.9 IF containment is NOT at atmospheric pressure, THEN do the following:

_______ a. Open 1-HV-MOV-102, CONT PURGE RELIEF VALVE, to raise the containment pressure to atmospheric.

_______ b. WHEN containment AND atmospheric pressure are equalized as indicated on 1-PI-LM-100A, B, C or D, THEN close 1-HV-MOV-102, CONT PURGE RELIEF VALVE.

_______ 5.1.10 Ensure the switch for 1-HV-3A/3B, Containment Iodine Filter Fans, is in the OFF position.

5.1.11 IF Mode 6 entry is anticipated, THEN do the following: **(References 2.3.4 and 2.3.7)**

_______ a. Notify the I&C Department to perform the section of ICP-RMS-1-RM-159, RMS-159 Containment Particulate Radiation Monitor Calibration, for Rotameter Calibration for Operation.

_______ b. Enter in the Action Statement Log to verify that 1-RM-RMS-159 and 1-RM-RMS-160 are operable with a sample flow rate of between 8 cfm and 12 cfm prior to movement of recently irradiated fuel within the containment per TRM 3.3.7.

NOTE: HP Release Form must be obtained prior to proceeding with this OP.

NOTE: Purge shall go through filters unless written authorization from Health Physics states otherwise.

_______ 5.1.12 Align containment purge through the iodine filters or to bypass the filter as per 0-OP-21.5, Operation of Auxiliary Building Iodine Filters.

5.0 INSTRUCTIONS

5.1 Directing Ventilation Exhaust Flow Through The Auxiliary Building Iodine Filter Banks

_______ 5.1.1 Verify Initial Conditions are satisfied.

_______ 5.1.2 Review Precautions and Limitations.

_______ 5.1.3 Place a check (✓) before each area to be placed through the filter bank(s):

_____ Auxiliary Building Central
_____ Auxiliary Building General
_____ Fuel Building
__✓__ Unit 1 Containment Purge
_____ Unit 2 Containment Purge
_____ Decon Building
_____ Unit 1 SFGDS
_____ Unit 2 SFGDS

_______ 5.1.4 IF the Auxiliary Building Iodine Filters MUST be placed in service for reasons such as to mitigate a radiological event to an area during or following any condition listed below, THEN enter the action of Tech Spec 3.7.12.A for one PREACS Train inoperable due to Tech Spec SR 3.7.12.3 / Tech Spec 5.5.10: **(References 2.4.2 and 2.4.3)**

- Painting
- Fire
- Chemical Release

5.1.5 IF the Auxiliary Building Iodine Filters are being placed in service for routine evolutions to an area during or following painting, THEN do the following:

N/A

a. Contact the Work Control Center to determine if painting is occurring in the area that will be aligned to the Auxiliary Building Iodine Filters.

b. Prior to continuing with this procedure, have the System Engineer complete Attachment 3 of VPAP-0904 to ensure continued operability of the charcoal filter.

NOTE: IF Auxiliary Building Central or General Exhaust Fans are to be aligned through the Aux Building Iodine Filter Banks for maintenance on dampers only (not during draining/venting operations), THEN Aux Building Roof Ventilators may be run after obtaining permission from Health Physics.

5.1.6 IF Unit SRO determines that Auxiliary Building Roof ventilators need to be secured, THEN place the control switch for the following in OFF:

N/A

- 1-HV-F-11A
- 1-HV-F-11B
- 1-HV-F-11C
- 1-HV-F-11D

Dominion

Ventilation Zone Painting / Solvents Approval - North Anna

VPAP-0904 - Attachment 3 **Page 1 of 1**

Requested By (Signature)	Extension	Date
Work Order / Work Request / DCP Number		Date
Location	Component	

Note: Operations Shift Supervisor approval is required each shift before initiation of work activities.

Auxiliary Building Ventilation Exhaust Filter (Mark N/A if not applicable)

Mark Number(s) of Affected Filter(s)		
Type of Primer	Weight of Primer (lb/gal)	Percent of VOC or Weight VOC
Type of Paint	Weight of Paint (lb/gal)	Percent of VOC or Weight VOC
Type of Solvent	Weight of Solvent	Percent of VOC
Maximum Volume of Paint / Solvents Allowed Without Affecting Filter Operability		
System Engineering Approval (Signature)		

Control Room Emergency Filter (Mark N/A if not applicable)

Mark Number(s) of Affected Filter(s)		
Type of Primer	Weight of Primer (lb/gal)	Percent of VOC or Weight VOC
Type of Paint	Weight of Paint (lb/gal)	Percent of VOC or Weight VOC
Type of Solvent	Weight of Solvent	Percent of VOC
Maximum Volume of Paint / Solvents Allowed Without Affecting Filter Operability		
System Engineering Approval (Signature)		

Operations Approval to Paint

Operations Shift Manager (Signature)	Date

Paint / Solvent Use Summary - Provide a copy of this form to System Engineering at the completion of each shift of coating.

Actual Volume of Primer Used	
Actual Volume of Paint Used	
Actual Volume of Solvent Used	
System Engineering Review (Signature)	Date

KEY: VOC-Volatile Organic Compounds

INFORMATION ONLY

VIRGINIA POWER

Engineering Transmittal

STD-GN-0041

To: Jim Dauberman, Operations

From: Lee Warnick, Plant Auxiliary Systems Engineering

Date: August 20, 1999 QA Category: SR

ET SE-99-046, REV. 0
TERMS USED IN TECH SPEC SECTIONS 4.6.4.3.B, 4.7.7.1.B, & 4.7.8.1.B
REGARDING CHARCOAL FILTERS
NORTH ANNA POWER STATION, UNIT 1 AND 2

Source Document

TECHNICAL SPECIFICATION SECTIONS 4.6.4.3.b, 4.7.7.1.b, & 4.7.8.1.b

References

1. UFSAR Table 6.2-51, expanatory notes for positions C.5.c, C.5.d, & C.6.b
2. Regulatory Guide (RG) 1.52, Rev. 2, dated 3/78, positions C.5.c, C.5.d, & C.6.b
3. "Evaluation and Control of Poisoning of Impregnated Carbons Used for Organic Iodide Removal," J. L. Kovach & L. Rankovic, 15th DOE Nuclear Air Cleaning Conference, 1978, p. 368-378.
4. "Effects on the Efficiency of Activated Carbon on Exposure to Welding Fumes," D. Gosh, NUREG/CP-0141 CONF-940738, 23rd DOE/NRC Nuclear Air Cleaning and Treatment Conference, July 25-28, 1994, p. 639-655.
5. "Effects of Welding Fumes on Nuclear Air Cleaning System Carbon Adsorber Banks," P. W. Robertson, NUREG/CP-0153 CONF-960715, 24th DOE/NRC Nuclear Air Cleaning and Treatment Conference, July 15-18, 1996, p. 525-533.
6. Letter from J. N. Donohew (NRC) to J. G. Dewease (Entergy Operations), "Interpretation of Filtration Unit Frequency of Testing Requirements Specified in the Technical Specifications and Regulatory Guide 1.52," dated 9/11/97.
7. Memo from R. M. Garver, II, to SNSOC, "Surveillance Requirement Position #11, Painting, Fire, or Chemical Release in Areas with Filter Systems," dated 11/5/92

Purpose of the ET

The Technical Specification (TS) sections referenced as source documents require testing of charcoal filters "following painting, fire, or chemical release in any ventilation zone communicating with the system." The purpose of this ET is to provide technical input to Operations regarding the bases for the terms "fire," "chemical release," and "communicating."

Discussion

Background. The Technical Specification (TS) sections referenced as source documents require testing of charcoal filters "following painting, fire, or chemical release in any ventilation zone communicating with the system." This wording echoes the requirement of RG 1.52 (reference 2), although NAPS is not specifically committed to the RG sections (reference 1). The basis for the RG wording is not explicitly stated. The NRC has recognized that "the terms 'painting,' 'fires,' 'chemical release,' and 'communicating' in RG 1.52 are subject to interpretation, and licensees are expected to develop interpretations of these terms to limit the HEPA and charcoal filter testing to situations which have a potential for degrading the ESF filtration system efficiency" (reference 6). Reference 3 is one of the earliest industry papers examining the negative effect of reduced charcoal efficiency caused by exposure to organic solvents (e.g., paint). The paper states that exposure of adsorbents to solvents such as paint fumes, degreasing and cleaning solvents, and welding fumes should be reduced through administrative controls.

Welding fumes. Subsequent research by two utilities has indicated that welding fumes do NOT have an effect on efficiency of the charcoal adsorbers (references 4 & 5). This research was based on creating excessive amounts of welding fumes from Type E308, E309, and E7018 welding electrodes and directing the fumes entirely through charcoal filter systems. Charcoal samples were then laboratory tested to the worst case test conditions (ASTM D3803-1989 at 30 C and 95% RH). No detrimental effects were observed on the efficiencies of the charcoal test samples. This research has been substantiated by subsequent experience at the two plants wherein charcoal filters were exposed to welding fumes, with no observable adverse impact on charcoal efficiency.

The welding electrodes studied above are representative of those commonly used at NAPS and are also considered to be representative of welding electrodes used infrequently at NAPS. For example, inconel welding rods (not used in the studies, but similar) have been used in the NAPS Auxiliary Building with no observable adverse impact on the process vent or other charcoal filters.

A concern noted in reference 4 is that welding fumes and smoke contain particulate matter (with average diameters from 0.3 to 0.7 microns). The study concluded, "any particulate that gains access to the plant exhaust system will eventually be trapped by either the prefilters or the upstream HEPA filters." NAPS has HEPA filters (with prefilters) associated with each charcoal filter assembly. Operator log 1-LOG-6D monitors differential pressure across the Auxiliary Building HEPA filters twice per shift to ensure that if the filter is operating, differential pressure remains less than a maximum value. The Control Room filters are not normally subject to exposure to welding fumes since they are isolated from the Turbine Building, except during 18-month 1(2)-PT-76.13A & B, and differential pressure across the HEPA filter is measured during this PT. Additional administrative controls are needed as identified in Required Action #2 below to ensure that the particulates in smoke do not impact the HEPA filters.

Cutting & Hot Work. Cutting, brazing, and other hot work activities can produce fumes or smoke in the plant, similar to welding. These activities do not have an effect on efficiency of charcoal filters, based on the following: a) cutting and brazing produce small amounts of smoke, b) smoke from these hot work activities are not expected to contain volatile organic compounds, c) the above evaluation of particulate matter in welding fumes would also bound the smoke from cutting or brazing, and d) these activities are administratively controlled.

Term: "Communicating." "Communicating" has been interpreted at NAPS (reference 7) "to mean that the ventilation system in the subject area is in operation and processing the atmosphere of the area" through the filter. This position implies that doors, dampers, and walls may be relied upon to separate areas and ventilation zones. For example, when the Fuel Building is lined up to the filters in Configuration "B" and the Auxiliary Building Central and General systems are in bypass mode, the Auxiliary Building is NOT communicating with the filters. A review of industry information indicates that this is consistent with industry practice. While these barriers may not be completely leak tight, the amount of leakage that the filters could be exposed to is insignificant relative to the volume of airflow in any system aligned to the filters. Additionally, past experience at NAPS substantiates that such activities in adjacent areas have no observable adverse impact on charcoal efficiency as measured by subsequent surveillance testing.

Conclusions

Relative to Technical Specifications for charcoal filters, the terms "fire" and "chemical release" exclude welding, cutting, brazing, and other hot work activities. The term "communicating" has been adequately defined in reference 7 to mean that the ventilation system in the subject area is in operation and processing the atmosphere of the area through the filter.

Required Actions

1. Operations procedures and periodic testing procedures will be revised in accordance with the attached CDS to delete "welding" from existing Precautions about charcoal filter degradation.
2. Procedures will be revised in accordance with the attached CDS to require verification of the HEPA filter differential pressure at the completion of filter operations.
   A) 0-OP-21.5, Operation of Auxiliary Building Iodine Filters, should verify the HEPA filter DP when removing the iodine filter from service.
   B) 0-OP-21.7, Main Control Room and Relay Room Emergency Ventilation Operation, should verify the associated HEPA filter DP when removing a fan from service if the fan was on Turbine Building supply.

DCE Review: Larry Runyon 8/27/99

Prepared by: L. T. Warnick 8/27/99

Reviewed by: M. G. Morgan 8/27/99

Approved by: R. D. McWhorter 8/27/99

☒ Activity Screening Checklist or ☐ Safety Evaluation included as Attachment 1 or ☐N/A
☒ PRC reviewed, no impact or ☐ PRC and PRCS included as Attachment
☐ CDS reviewed, no impact or ☒ CDS included as Attachment 2

Additional Attachments

None.

Copy to:

See Distribution List

VIRGINIA POWER

INFORMATION ONLY

Activity Screening Checklist

Attachment 1, E.T. SE 99-046,
Revision 0

1. Identification of Governing Document	2. Applicable Station	3. Applicable Unit
E.T. SE 99-046	[X] North Anna Power Station [ ] Surry Power Station	[X] Unit 1 [X] Unit 2

4. Brief Description of the Activity

The purpose of this ET is to provide technical input to Operations regarding the bases for the terms "fire," "chemical release," and "communicating." Relative to Technical Specifications for charcoal filters, the terms "fire" and "chemical release" exclude welding, cutting, brazing, and other hot work activities. The term "communicating" has been defined.

5. General Screening (Definitions are provided in VPAP-3001.)

Question	Yes	No
A. Does this activity require a change to the Operating License or Technical Specifications?	[ ] Yes	[X] No
B. Does this activity alter (temporarily or permanently) the information, design, function, ability to function, or method of performing the function of a structure, system, or component as described in the SAR?	[ ] Yes	[X] No
C. Does this activity modify a procedure or method of operation as described, outlined or summarized in the SAR?	[ ] Yes	[X] No
D. Does this activity perform a test or experiment that is not described in the SAR?	[ ] Yes	[X] No
E. Does this activity involve a change to the Environmental Protection Plan, or a change, test, or experiment that may affect the environment?	[ ] Yes	[X] No
F. Does this activity involve a temporary modification?	[ ] Yes	[X] No

6. Discussion and References

This clarification of specific terms in Technical Specifications is not altering the information in the SAR. Welding would not normally be included in the meaning of the terms "fire" or "chemical release." SNSOC agreed that "Surveillance Requirement Position #11" regarding the term "communicating" is not considered a Tech Spec Interpretation because it is consistent with existing regulatory guidance, and is therefore not altering the information in the SAR.

The procedures modified by these clarifications are not discussed in the SAR. This is not a test or experiment or a temporary modification.

Note: If Any Response is "Yes," a Safety Evaluation Must be Performed In Accordance With VPAP-3001, Safety Evaluations.

7. Preparer Name (Please Print) L. T. Warnick	8. Title System Engineer	
9. Preparer Signature [signature]		10. Date 8/27/99
11. Reviewer Name (Only If Non-Authorized Preparer-Please Print)	12. Title	
13. Reviewer Signature		14. Date

Key: SAR-Safety Analysis Report

3.7 PLANT SYSTEMS

3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.12 Two ECCS PREACS trains shall be OPERABLE.

------------------------------ NOTE ------------------------------
The ECCS pump room boundary openings not open by design may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS PREACS train inoperable.	A.1 Restore ECCS PREACS train to OPERABLE status.	7 days
B. Two ECCS PREACS trains inoperable due to inoperable ECCS pump room boundary.	B.1 Restore ECCS pump room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.12.1 Operate each ECCS PREACS train for $\geq$ 10 continuous hours with the heaters operating.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.2	Actuate each ECCS PREACS train by aligning Safeguards Area exhaust flow and Auxiliary Building Central exhaust flow through the Auxiliary Building HEPA filter and charcoal adsorber assembly.	31 days
SR 3.7.12.3	Perform required ECCS PREACS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.4	Verify Safeguards Area exhaust flow is diverted and each Auxiliary Building filter bank is actuated on an actual or simulated actuation signal.	18 months
SR 3.7.12.5	Verify one ECCS PREACS train can maintain a negative pressure relative to adjacent areas during post accident mode of operation.	18 months on a STAGGERED TEST BASIS

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in general conformance with the frequencies and requirements of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

a. (continued)

with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)	1000 ± 10% cfm
Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
MCR/ESGR EVS	1000 ± 10% cfm
ECCS PREACS	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than the

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

c. (continued)

value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity specified below.

ESF Ventilation System	Penetration	RH
MCR/ESGR EVS	2.5%	70%
ECCS PREACS	5%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
MCR/ESGR EVS	4 inches W.G.	1000 ± 10% cfm
ECCS PREACS	5 inches W.G.	≤ 39,200 cfm

e. Demonstrate that the heaters for each of the ESF systems dissipate ≥ the value specified below when tested in accordance with ASME N510-1975.

ESF Ventilation System	Wattage
MCR/ESGR EVS	3.5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or

(continued)

10. 028AA2.12 085/NEW//H/4/2.7/4.1/5/

Unit 1 is at 100% power with the PRZR Level Channel Defeat Switch selected to position 461/460.

An instrument malfunction causes the following plant response:

- Letdown automatically isolated.
- Normal charging flow control valve 1-CH-FCV-1122 fully open.

Which ONE of the following identifies the failed channel, **AND** includes the protection function described by the Technical Specification Bases that is impacted by this failure?

A. 1-RC-LT-1460 is failed ; primary protection against RCS overpressurization.

B. 1-RC-LT-1460 is failed ; protection against water relief through the PRZR Safety Valves.

C. 1-RC-LT-1461 is failed ; primary protection against RCS overpressurization.

D. 1-RC-LT-1461 is failed ; protection against water relief through the PRZR Safety Valves.

a. Incorrect. First part is incorrect but plausible since the candidate may not have a solid understanding of switch function and failure modes and effects. Second part is incorrect but plausible since a large load rejection would cause a large insurge which could result in overpressurization, however as described in the bases the hi water level trip is backup protection.

b. Incorrect. First part incorrect but plausible as noted above. Second part is correct IAW TS bases 3.3.1 function 9.

c. Incorrect. First part is correct; based on the conditions giventhe controling channel (1461 in this case) has failed low. Second part incorrect but plausible as discussed in Distractor a.

d. Correct. First part is correct as discussed in Distractor c. Second part is also correct as explained in the TS bases this protection is needed for the safety valves which must pass steam to achieve their design energy removal rate. However, the candidate who does not have an in depth knowledge of the bases, may consider water relief as merely an "additional benefit" of the trip function, and thus discount this answer.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Pressurizer (PZR) Level Control Malfunction

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions:
(CFR: 43.5 / 45.13)

Cause for PZR level deviation alarm: controller malfunction or other instrumentation malfunction

Tier: 1
Group: 1

Technical Reference: drawing 108D014 sh. 4 of 17 & 5655D33 sh. 11 of 16, and TS 3.3.1 bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Pressurizer Pressure (continued)

b. Pressurizer Pressure—High

The Pressurizer Pressure—High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires three channels of the Pressurizer Pressure—High to be OPERABLE.

The Pressurizer Pressure—High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure—High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure—High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level—High

distracter, as stated this is "backup" not "primary" as given in ~~#~~ distracter

The Pressurizer Water Level—High trip Function provides a backup signal for the Pressurizer Pressure—High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level—High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection

Answer

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Pressurizer Water Level—High (continued)

interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

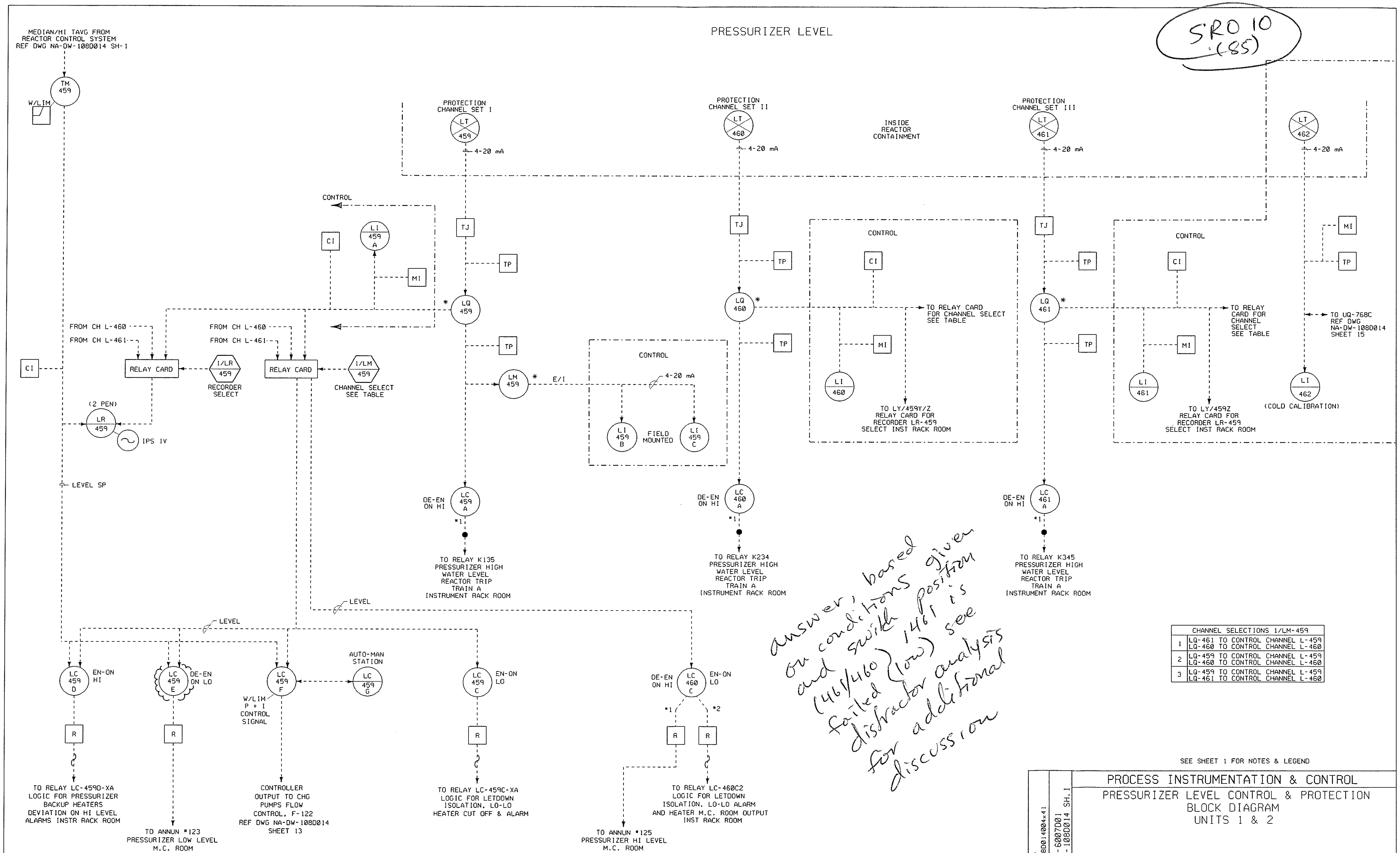
In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level—High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow—Low

The Reactor Coolant Flow—Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, which is approximately 30% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE in MODE 1 above P-7.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core because of the higher power level. In MODE 1 below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.



SRO 10 (85)

answer, based on conditions given and switch position (461/460) L461 is failed (low) see distractor analysis for additional discussion

CHANNEL SELECTIONS I/LM-459	
1	LQ-461 TO CONTROL CHANNEL L-459 LQ-460 TO CONTROL CHANNEL L-460
2	LQ-459 TO CONTROL CHANNEL L-459 LQ-460 TO CONTROL CHANNEL L-460
3	LQ-459 TO CONTROL CHANNEL L-459 LQ-461 TO CONTROL CHANNEL L-460

SEE SHEET 1 FOR NOTES & LEGEND

PROCESS INSTRUMENTATION & CONTROL
PRESSURIZER LEVEL CONTROL & PROTECTION BLOCK DIAGRAM UNITS 1 & 2

REF DWG NA-DW-6007D01 NA-DW-108D014 SH.1

CAD NO: SARAH15 C:\usr\dgn\N108D014004x41

Dominion — NUCLEAR ENGINEERING — NORTH ANNA POWER STATION

REVISED PER DCR 2002-0952. THIS DWG SUPERSEDES THE REV 1 ORIGINAL. — SCC

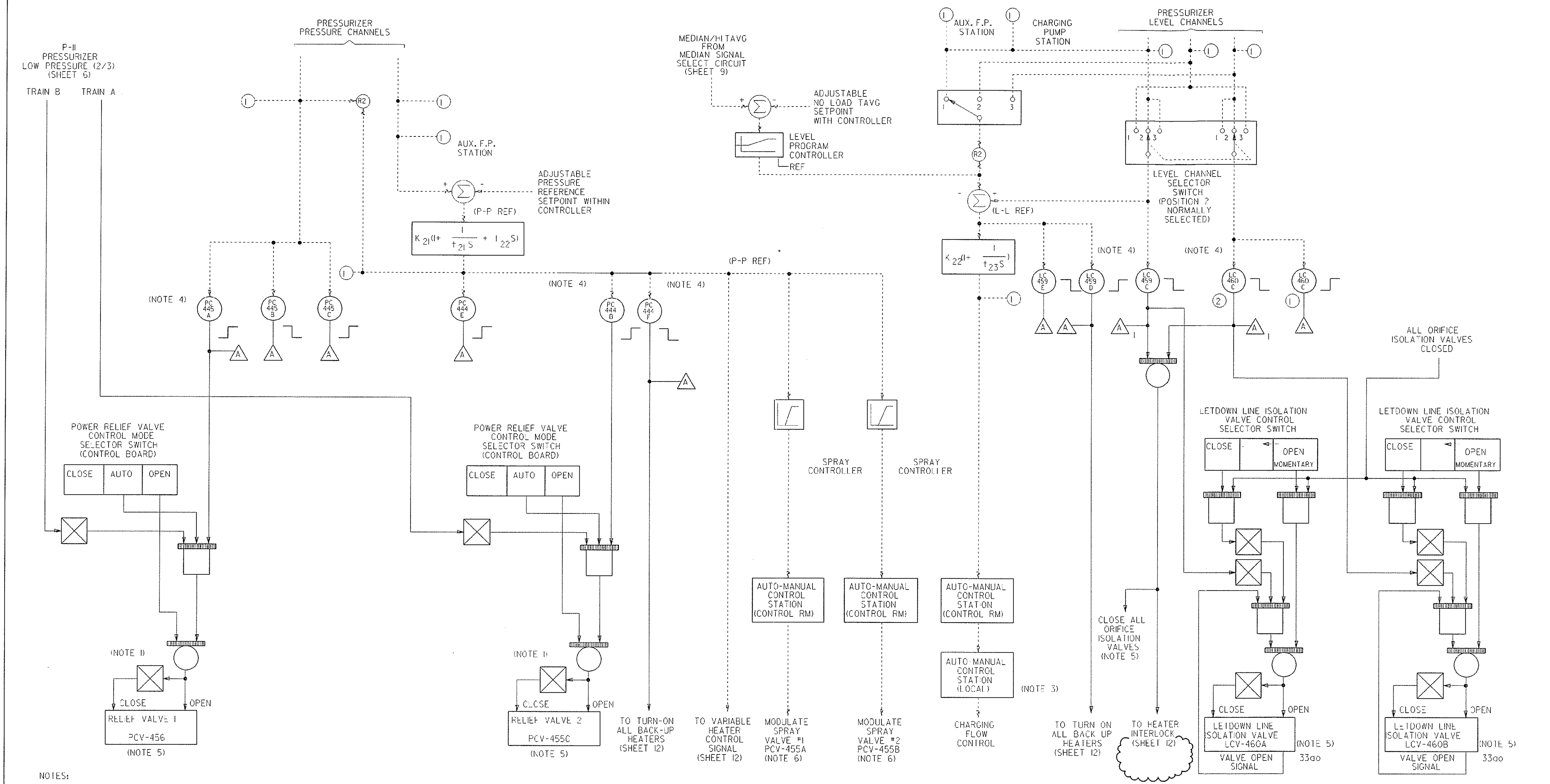
REVISION DESCRIPTION — DSGN

DRAWING NUMBER NA-DW-108D014 SH 4 OF 17 REVISION 2

PC=N01

29-AUG-2002 08:31

PRIOR TO USING FOR DESIGN WORK CHECK DMIS FOR WORK PENDING



NOTES:

1. LOGIC OUTPUT OPERATES 2 SOLENOID VENT VALVES IN SERIES TO INTERLOCK THE AIR LINE TO EACH VALVE DIAPHRAGM. THE SOLENIOD VALVES ARE DE-ENERGIZED TO VENT, CAUSING THE MAIN RELIEF VALVE TO CLOSE IN 2 SECONDS.
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
3. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
4. PRESSURE BISTABLES PC-444B, PC-444F AND PC-445A ARE 'ENERGIZE TO ACTUATE'.
5. OPEN/SHUT INDICATION IN CONTROL ROOM.
6. A LIGHT SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN IT IS NOT FULLY CLOSED.

NUCLEAR STEAM SUPPLY SYSTEM

FUNCTIONAL DIAGRAM
PRESSURIZER PRESSURE & LEVEL CONTROL
UNITS 1 & 2

VIRGINIA POWER
NORTH ANNA POWER STATION

REVISED PER DCR 2001-1268 THIS DWG SUPERSEDES REV 1 ORIGINAL	ELC	
REVISION DESCRIPTION	DSGN	

CAD NO: COOPERE
C:\usr\dgn\N5655D3301l1oo1
REF DWG NA-DW-5655D33 SH.J

DRAWING NUMBER: NA-DW-5655D33 SH 11 OF 16
REVISION: 2

PC=ND1

12-SEP-2001 06:18

PRIOR TO USING FOR DESIGN WORK CHECK DMIS FOR WORK PENDING

11. 035A2.02 086/BANK//H/3/4.2/4.4/4/

Given the following conditions:

- Unit 1 was initially at 100% power.
- A loss of offsite power occurs.
- 1-MS-PCV-101A, "A" SG PORV, sticks fully open.
- The crew was able to locally override 1-MS-PCV-101A closed.

The plant has been stabilized, and the crew is continuing recovery actions in accordance with 1-ES-0.1, Reactor Trip Response.

Which ONE of the following identifies the current status of 1-MS-PCV-101A **AND** the LCO for TS 3.7.4, Steam Generator Power Operated Relief Valves?

A. 1-MS-PCV-101A is operable and LCO 3.7.4 is met.

B. 1-MS-PCV-101A is inoperable and must be restored to operable prior to startup.

C. 1-MS-PCV-101A is inoperable and must be restored to operable status within 24 hours.

D. 1-MS-PCV-101A is inoperable and must be restored to operable status within 7 days.

a. Correct. As described in the bases since the valve is capable of being cycled fully open and closed in a controlled manner locally it is still considered operable (see TS bases 3.7.4), with all three operable the LCO is met.

b. Incorrect. As noted above the valve is operable, but if the candidate erroneously assumes it is inoperable they could still pick this distractor since it is a provision of TS 3.0.4.

c. Incorrect. As noted above the valve is operable, but if the candidate erroneously assumes it is inoperable they could still pick this distractor since it is a TS 3.7.4 action time.

d. Correct. As noted above the valve is operable, but if the candidate erroneously assumes it is inoperable they could still pick this distractor since it is the TS 3.7.4 action time for one SG PORV inoperable.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Steam Generator System (S/GS)

Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.5)

Reactor trip/turbine trip

Tier: 2
Group: 2

Technical Reference: TS 3.7.4 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

B 3.7 PLANT SYSTEMS

B 3.7.4 Steam Generator Power Operated Relief Valves (SG PORVs)

BASES

BACKGROUND

The SG PORVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the condenser dump valves not be available, as discussed in the UFSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the emergency condensate storage tank (ECST) (or, alternately, with main feedwater from the condenser hotwell or main condensate tanks, if available).

One SG PORV line for each of the three steam generators is provided. Each SG PORV line consists of one SG PORV and an associated upstream manual isolation valve.

also see pg 3.7.4-3

The SG PORVs are provided with upstream manual isolation valves to permit their being tested at power, and to provide an alternate means of isolation. The SG PORVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The SG PORVs are provided with a backup supply tank which is pressurized from the instrument air header via a check valve arrangement that, on a loss of pressure in the normal instrument air supply, automatically supplies air to operate the SG PORVs. The air supply is sized to provide the sufficient pressurized air to operate the SG PORVs until manual operation of the SG PORVs can be established.

answer (specific knowledge of this bases is required to get correct answer)

A description of the SG PORVs is found in Reference 1. The SG PORVs are OPERABLE when they are capable of providing controlled relief of the main steam flow and capable of being fully opened and closed, either remotely or by local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the SG PORVs is established by the capability to cool the unit to RHR entry conditions. The SG PORVs are used in conjunction with auxiliary feedwater supplied from the ECST (or, alternately, with main feedwater from the condenser hotwell or main condensate tanks, if

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

available). Adequate inventory is available in the ECST to support operation for 2 hours in MODE 3 followed by a 4 hour cooldown to the RHR entry conditions.

In the SGTR accident analysis presented in Reference 2, the SG PORVs are assumed to be used by the operator to cool down the unit to RHR entry conditions when the SGTR is accompanied by a loss of offsite power, which renders the condenser dump valves unavailable. Prior to operator actions to cool down the unit, the SG PORVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event. Thus, the SGTR is the limiting event for the SG PORVs. The requirement for three SG PORVs to be OPERABLE satisfies the SGTR accident analysis requirements, including consideration of a single failure of one SG PORV to open on demand.

The SG PORVs are equipped with manual isolation valves in the event an SG PORV spuriously fails open or fails to close during use.

The SG PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Three SG PORV lines are required to be OPERABLE. One SG PORV line is required from each of three steam generators to ensure that at least one SG PORV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second SG PORV line on an unaffected steam generator. The manual isolation valves must be OPERABLE to isolate a failed open SG PORV line. A closed manual isolation valve does not render it or its SG PORV line inoperable because operator action time to open the manual isolation valve is supported in the accident analysis.

(continued)

BASES

LCO (continued)

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Dump System.

key aspect of operability

An SG PORV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing, remotely or by local manual operation on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the SG PORVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required SG PORV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV lines, a nonsafety grade backup in the Steam Dump System, and MSSVs.

B.1

With two or more SG PORV lines inoperable, action must be taken to restore all but one SG PORV line to OPERABLE status. Since the upstream manual isolation valve can be closed to isolate an SG PORV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable SG PORV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV lines.

C.1 and C.2

If the SG PORV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the SG PORVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the SG PORVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an SG PORV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the upstream manual isolation valve is to isolate a failed SG PORV. Cycling the upstream manual isolation valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the upstream manual isolation valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.

2. UFSAR, Section 15.4.3.

3.7 PLANT SYSTEMS

3.7.4 Steam Generator Power Operated Relief Valves (SG PORVs)

LCO 3.7.4 Three SG PORV lines shall be OPERABLE.

still require 3 post-trip (mode 3)

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SG PORV line inoperable.	A.1 Restore required SG PORV line to OPERABLE status.	7 days
B. Two or more required SG PORV lines inoperable.	B.1 Restore all but one SG PORV line to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND	6 hours
	C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	24 hours

distractor
c, d

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify one complete cycle of each SG PORV.	18 months
SR 3.7.4.2 Verify one complete cycle of each SG PORV manual isolation valve.	18 months

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

distractor

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specific condition in the Applicability for an unlimited period of time,

not true for case where a PORV is inop

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

b. After performance of a risk evaluation, consideration of the results, determination of the acceptability of the MODE change, and establishment of risk management actions, if appropriate, or not done yet

c. When a specific value or parameter allowance has been approved by the NRC. no true for PORV

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

12. 036AG2 .4.8 087/NEW//H/4/3.8/4.5/8/

Given the following conditions:

- Unit 2 is at 100% power with Unit 1 core off-load in progress.
- A fuel assembly separates from its top nozzle while being withdrawn from the core.
- The crew enters 0-AP-30, Fuel Failure During Handling.

While the crew is performing time-critical actions of 0-AP-30, the Unit 2 reactor trips due to accidental jarring of a reactor trip breaker.

Which ONE of the following identifies the requirements for prioritization of 0-AP-30 and Unit 2 EOPs (2-E-0, Reactor Trip or Safety Injection, and 2-ES-0.1, Reactor Trip Response)?

A. Regardless of manpower availability, the crew should suspend performance of 0-AP-30 while performing the immediate actions of 2-E-0, and then perform 0-AP-30 in parallel with Unit 2 EOPs.

B. Regardless of manpower availability, the crew should suspend performance of 0-AP-30 while performing 2-E-0, and should not resume performing 0-AP-30 until Unit 2 is stable per 2-ES-0.1.

C. The crew should complete the time-critical actions of 0-AP-30 prior to entering 2-E-0, and then perform Unit 2 EOPs in parallel with 0-AP-30.

D. If manpower permits, the crew should continue performing the time-critical actions of 0-AP-30 while performing the immediate actions of 2-E-0, and then perform 0-AP-30 in parallel with Unit 2 EOPs

a. Incorrect. Plausible since EOPs typically take presidence however the procedure provides for concurrent performance on not-to-interfere with basis so since manpower is not an issue this distractor is incorrect.

b. Incorrect. plausible as noted above; if sufficeient manpower is available there is no requirement to suspend performance of the AP until the unit is stabilized.

c. Incorrect. As discussed above EOPs typically take presidence and while there is a need to complete time-critical actions within the specified time frame, verification of the reactor trip (2-E-0) although not listed in GOP-17 for time critical, contains immediate actions which by definition are required to be implemented without delay; thus 2-E-0 entry cannot be delayed by virtue of performing actions of a subbordinate procedure.

d. Correct. As given, if there is sufficient manpower to allow performance of a subordinate procedure without interfering with EOP performance, then that method is preferred since that method is benificial in ensuring there is no discontinuity in execution of the subordinate procedure.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Fuel Handling Incidents

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
(CFR: 41.10 / 43.5 / 45.13)

Tier: 1
Group: 2

Technical Reference: OP-AP-104 & 0-AP-30

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

K/A ~~[illegible]~~ 036AG2.4.8

3.7 Abnormal Operating Procedures

> **NOTE:** Except at Millstone, immediate actions are identified clearly as such in Abnormal Operating Procedures.

Procedure User

3.7.1 **PERFORM** immediate actions from memory.

3.7.2 **APPLY** the following communication techniques when using Abnormal Operating Procedures:

- A NOTE or CAUTION is read verbatim when encountered for the first time during an event
- If not applicable, a NOTE or CAUTION need not be verbalized
- If encountered repeatedly, a NOTE or CAUTION may be paraphrased
- A NOTE or CAUTION may be directed to applicable crew members; in such cases, the applicable crew member will paraphrase or acknowledge the NOTE or CAUTION

> **NOTE:** The priority of procedures depends upon the events in progress. Some abnormal operating procedures must be implemented while emergency operating procedures are in effect. In cases of parallel procedure usage, the EOP receives priority and immediate actions are completed before parallel procedure usage. When using an AOP in parallel with the EOP, only those steps in the AOP that ensure success of the EOP are required to be performed.

3.7.3 **REVIEW** the immediate action steps of abnormal operating procedures after performance to ensure required actions have been taken.

3.7.4 **COMPLETE** subsequent action steps of the abnormal operating procedures as listed in the procedures.

3.7.5 **IF** a variance becomes necessary during performance of abnormal operating procedures, **THEN** **PERFORM** the following:

a. Obtain concurrence from a second SRO, if possible.

Dominion

NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
0-AP-30	FUEL FAILURE DURING HANDLING (WITH TWO ATTACHMENTS)	12
		PAGE 1 of 13

PURPOSE

To provide instructions to follow in the event of a fuel failure during handling.

NO INITIAL ACTIONS

ENTRY CONDITIONS

This procedure is entered when any of the following conditions exist:

- Gas bubbles or discoloration of the water in the area of a fuel assembly, or
- Report of a fuel failure from refueling personnel, or
- Increasing radiation level on any of the following:
    - 1-RM-RMS-159, Unit 1 Containment Gaseous
    - 1-RM-RMS-160, Unit 1 Containment Particulate
    - 1-RM-RMS-162, Unit 1 Manipulator Crane
    - 2-RM-RMS-259, Unit 2 Containment Gaseous
    - 2-RM-RMS-260, Unit 2 Containment Particulate
    - 2-RM-RMS-262, Unit 2 Manipulator Crane
    - 1-RM-RMS-152, New Fuel Storage Area
    - 1-RM-RMS-153, Spent Fuel Pit Bridge Crane

COMMON

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	NOTIFY CONTROL ROOM	

NOTE: The Bases for Tech Spec 3.7.10 and 3.7.11 define "recently irradiated fuel" as "fuel that has occupied part of a critical reactor core within the previous 300 hours".

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.___	CHECK IF MANUAL CONTROL ROOM BOTTLE AIR DUMP AND ISOLATION IS REQUIRED:	
	☐ a) Movement of recently irradiated fuel - IN PROGRESS	☐ a) IF SRO desires Control Room bottle air dump and isolation, THEN continue with AER Step 2b.
	b) Either of the following - SATISFIED:	☐ b) GO TO Step 3.
	☐ • Verbal report of a fuel handling accident	
	OR	
	☐ • Any containment radiation monitor - IN ALARM	

(STEP 2 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	CHECK IF MANUAL CONTROL ROOM BOTTLE AIR DUMP AND ISOLATION IS REQUIRED: (Continued)	
	c) Place the following CONTROL RM BOTTLED AIR SYS PANEL control switches in - OPEN:	
	☐ • 1-HV-SOV-1300A 1-HV-SOV-1308A - PANEL 1A, 1-EI-CB-156A	
	☐ • 1-HV-SOV-1300B - PANEL 1B, 1-EI-CB-156B	
	☐ • 1-HV-SOV-1300C 1-HV-SOV-1308B - PANEL 2A, 2-EI-CB-156A	
	☐ • 1-HV-SOV-1300D - PANEL 2B, 2-EI-CB-156B	
	☐ • 2-HV-SOV-2300A - PANEL 1A, 1-EI-CB-156A	
	☐ • 2-HV-SOV-2300B 2-HV-SOV-2308A - PANEL 1B, 1-EI-CB-156B	
	☐ • 2-HV-SOV-2300C - PANEL 2A, 2-EI-CB-156A	
	☐ • 2-HV-SOV-2300D 2-HV-SOV-2308B - PANEL 2B, 2-EI-CB-156B	
	d) Verify Control Room Fan Status:	
	☐ 1) 1-HV-F-15 - OFF	☐ 1) Stop 1-HV-F-15
	☐ 2) 1-HV-F-41 - ON	☐ 2) Start 1-HV-F-41
	☐ 3) 2-HV-F-41 - ON	☐ 3) Start 2-HV-F-41

(STEP 2 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	CHECK IF MANUAL CONTROL ROOM BOTTLE AIR DUMP AND ISOLATION IS REQUIRED: (Continued)	
	e) Verify fan status on Auxiliary Shutdown Panels:	☐ e) Start fans.
☐	• 1-HV-F-42 - ON	
☐	• 2-HV-F-42 - ON	
	f) Verify Control Room Damper status:	☐ f) Close dampers.
☐	• 1-HV-AOD-161-1 - CLOSED	
☐	• 1-HV-AOD-160-1 - CLOSED	
☐	• 1-HV-AOD-161-2 - CLOSED	
☐	• 1-HV-AOD-160-2 - CLOSED	

NOTE: If electrical power is unavailable to the Manipulator Crane, then manual operation may be required.

3.____ PLACE THE FAILED FUEL ASSEMBLY IN A SAFE LOCATION

4.____ EVACUATE AFFECTED AREA

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: For a Fuel Handling Accident in Containment, the required closure actions will be based on the existing conditions and established radiation protection practices. When acceptable radiological protection conditions exist for closure team personnel, containment closure will be established within 45 minutes.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.___	CHECK ACCIDENT - IN FUEL BUILDING	Do the following: ☐ a) Initiate ATTACHMENT 2, CLOSING AIRLOCK DOORS IN AN EMERGENCY for affected Containment. b) Verify Containment Closure: 1) Refer to the applicable Containment Boundary Breach Log, for open penetrations: ☐ • 1-LOG-18 ☐ • 2-LOG-18 ☐ 2) Notify Health Physics. ☐ 3) Establish containment closure for open penetrations.
6.___	NOTIFY HEALTH PHYSICS DEPARTMENT	
7.___	CHECK IF ACCIDENT IS IN UNIT 1 CONTAINMENT	☐ GO TO Step 10.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. ___	CHECK UNIT 1 CONTAINMENT RADIATION MONITORS - ANY HIGH RADIATION ALARM EXISTS:	☐ IF a high radiation alarm is received, THEN RETURN TO Step 2.
	☐ • 1-RM-RMS-159	☐ GO TO Step 16.
	OR	
	☐ • 1-RM-RMS-160	
	OR	
	☐ • 1-RM-RMS-162	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: A HI-RAD alarm on 1-RM-RMS-159, 1-RM-RMS-160, or 1-RM-RMS-162 will trip Containment Purge Supply and Exhaust Fans to Unit 1 Containment unless Unit 2 Containment is being ventilated.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.___	VERIFY AUTOMATIC ACTUATIONS HAVE OCCURRED ON CONTAINMENT RADIATION MONITOR ALARM:	
	a) Containment Purge Fans - TRIPPED:	☐ a) IF Unit 2 Containment is being ventilated, THEN continue to operate Containment Purge Fans.
	☐ • 1-HV-F-4A Supply	
	☐ • 1-HV-F-4B Supply	
	☐ • 1-HV-F-5A Exhaust	
	☐ • 1-HV-F-5B Exhaust	
	b) Isolation dampers on Purge Supply and Exhaust - CLOSED:	☐ b) Close any open damper.
	☐ • 1-HV-MOV-100A	
	☐ • 1-HV-MOV-100B	
	☐ • 1-HV-MOV-100C	
	☐ • 1-HV-MOV-100D	
	☐ • 1-HV-MOV-101	
	☐ • 1-HV-MOV-102	
	☐ c) GO TO Step 16	
10.___	CHECK IF ACCIDENT IS IN UNIT 2 CONTAINMENT	☐ GO TO Step 13.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.___	CHECK UNIT 2 CONTAINMENT RADIATION MONITORS - ANY HIGH RADIATION ALARM EXISTS:	☐ IF a high radiation alarm is received, THEN RETURN TO Step 2.
	☐ • 2-RM-RMS-259	☐ GO TO Step 16.
	OR	
	☐ • 2-RM-RMS-260	
	OR	
	☐ • 2-RM-RMS-262	

NOTE: A HI-RAD alarm on 2-RM-RMS-259, 2-RM-RMS-260, or 2-RM-RMS-262 will trip Containment Purge Supply and Exhaust Fans to Unit 2 Containment unless Unit 1 Containment is being ventilated.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12.___	VERIFY AUTOMATIC ACTUATIONS HAVE OCCURRED ON CONTAINMENT RADIATION MONITOR ALARM:	
	a) Verify Containment Purge Fans - TRIPPED:	☐ a) IF Unit 1 Containment is being ventilated, THEN continue to operate Containment Purge Fans.
	☐ • 1-HV-F-4A Supply	
	☐ • 1-HV-F-4B Supply	
	☐ • 1-HV-F-5A Exhaust	
	☐ • 1-HV-F-5B Exhaust	

(STEP 12 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12.	VERIFY AUTOMATIC ACTUATIONS HAVE OCCURRED ON CONTAINMENT RADIATION MONITOR ALARM: (Continued)	
	b) Verify Isolation Dampers on Purge Supply and Exhaust - CLOSED:	☐ b) Close any open damper.
	☐ • 2-HV-MOV-200A	
	☐ • 2-HV-MOV-200B	
	☐ • 2-HV-MOV-200C	
	☐ • 2-HV-MOV-200D	
	☐ • 2-HV-MOV-201	
	☐ • 2-HV-MOV-202	
	☐ c) GO TO Step 16	
13.___	CHECK FUEL BUILDING RADIATION MONITORS - ANY HIGH RADIATION ALARM EXISTS:	☐ WHEN a high radiation alarm is received, THEN perform Step 14.
	☐ • 1-RM-RMS-152, NEW FUEL STORAGE	☐ GO TO Step 16.
	OR	
	☐ • 1-RM-RMS-153, SPENT FUEL PIT BRIDGE CRANE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: To prevent entry of contaminants into the Control Room, the Hi-Hi alarm on 1-RM-RMS-152 and 1-RM-RMS-153 should not be reset until Health Physics has verified that the Turbine Building atmosphere is acceptable.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.___	VERIFY CONTROL ROOM ISOLATION - ACTUATED:	
	a) Fuel Building Radiation Automatic Interlock key switch in - ENABLE	☐ a) IF SRO desires Control Room bottle air dump and isolation, THEN continue with Step 14b. IF NOT, THEN GO TO Step 15.
	b) Verify automatic dump of Bottled Air System - ON:	☐ b) Manually dump Control Room Control Room Bottled Air System.
☐	• 1-HV-PI-1311	
☐	• 2-HV-PI-2311	
	c) Verify Control Room Fan Status:	
☐	1) 1-HV-F-15 - OFF	☐ 1) Stop 1-HV-F-15
☐	2) 1-HV-F-41 - ON	☐ 2) Start 1-HV-F-41
☐	3) 2-HV-F-41 - ON	☐ 3) Start 2-HV-F-41
	d) Verify fan status on Auxiliary Shutdown Panels:	☐ d) Start Fans.
☐	• 1-HV-F-42 - ON	
☐	• 2-HV-F-42 - ON	

(STEP 14 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	VERIFY CONTROL ROOM ISOLATION - ACTUATED: (Continued)	
	e) Verify Control Room Damper status:	☐ e) Close dampers.
☐	• 1-HV-AOD-161-1 - CLOSED	
☐	• 1-HV-AOD-160-1 - CLOSED	
☐	• 1-HV-AOD-161-2 - CLOSED	
☐	• 1-HV-AOD-160-2 - CLOSED	
15. ___	VERIFY FUEL BUILDING VENTILATION - ALIGNED:	☐ Align Fuel Building ventilation through the charcoal filters.
	a) Fuel Building Exhaust Fan - AT LEAST ONE ON:	☐ GO TO Step 16.
☐	• 1-HV-F-7A	
	<u>OR</u>	
☐	• 1-HV-F-7B	
☐	b) Fuel Building Supply Fan 1-HV-F-6 - OFF	
	c) Check either set of Auxiliary Building Iodine Filter Dampers in FILTER:	
☐	• 1-HV-AOD-107A 1,2,3,4	
	<u>OR</u>	
☐	• 1-HV-AOD-107B 1,2,3,4	

(STEP 15 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15.	VERIFY FUEL BUILDING VENTILATION - ALIGNED: (Continued)	
	d) Fuel Building Exhaust aligned to charcoal filter 1-HV-FL-3A or 1-HV-FL-3B as indicated by the following dampers in FILTER:	
	☐ • 1-HV-AOD-107-1	
	☐ • 1-HV-AOD-107-2	
	☐ • 1-HV-AOD-107-3	
	☐ • 1-HV-AOD-107-4	
16.___	INITIATE EPIP-1.01, EMERGENCY MANAGER CONTROLLING PROCEDURE	
17.___	ALIGN THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM - WITHIN ONE HOUR OF EVENT:	
	a) Place ONE of the following Emergency Ventilation Fans in service on Turbine Building supply using 0-OP-21.7, Main Control Room and Relay Room Emergency Ventilation Operation:	
	☐ • 2-HV-F-41	
	☐ • 1-HV-F-42	
	☐ • 2-HV-F-42	
	☐ b) Verify all other available Emergency Ventilation fans are in service on recirculation	
18.___	CHECK IF CAUSE OF ACCIDENT - CORRECTED	☐ <u>WHEN</u> cause of accident has been corrected, <u>THEN</u> continue with Step 19.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.___	RESTORE VENTILATION SYSTEMS TO NORMAL USING: ☐ • 0-OP-21.7, MAIN CONTROL ROOM AND RELAY ROOM EMERGENCY VENTILATION OPERATION ☐ • 1-OP-21.2, CONTAINMENT PURGE ☐ • 2-OP-21.2, CONTAINMENT PURGE	
20.___	RETURN TO PROCEDURE IN EFFECT	

- END -

1. UFSAR Section 9.4.9
2. UFSAR Section 11.4.2.18
3. UFSAR Section 15.4.5
4. 11715-ESK-6KE
5. 11715-ESK-6KF
6. TRM 3.3.7
7. Tech Spec 3.7.10
8. TRM 3.9.5
9. Tech Spec 3.7.15
10. DCP 89-32
11. 1(2)-OP-21.2, CONTAINMENT PURGE
12. 0-OP-21.7, MAIN CONTROL ROOM AND RELAY ROOM EMERGENCY VENTILATION OPERATION
13. EPIP-1.01, EMERGENCY MANAGER CONTROLLING PROCEDURE
14. Tech Spec Amendment 198, Containment Building Penetrations Unit 1
15. Tech Spec Amendment 179, Containment Building Penetrations Unit 2
16. DR N-99-0215, Control Room Pressurization, Isolation, and Ventilation
17. DCP 00-169, Control Room Bottled Air System Modification
18. TSCR N-011, Alternate Source Team (Rev. 11, Step 2.a and note, 14.a and Attachment 2, Step 1)

19. Tech Spec change N-052A, Deletion of MCR/ESGR Bottled Air System from Tech Specs (Rev. 12, Step 17)

1. ___ Have the Maintenance Department install the Equipment Door and Temporary Penetration Plate using 0-MCM-1204-05, EMERGENCY INSTALLATION OF EQUIPMENT DOOR AND TEMPORARY PENETRATION PLATE, as directed by Health Physics based on radiological conditions.

* *

CAUTION: At least one Personnel Airlock door must be closed. The inner door is the preferred door. If possible, then both doors should be closed.

* *

2. ___ Ensure any covers placed over the door seating surfaces are removed.

___ a) Close the Personnel Airlock Door as follows:

___ b) Verify door seating surfaces are clean.

c) Coordinate with Control Room to manipulate ventilation as necessary to equalize pressure between the Auxiliary Building and Containment:

___ • Containment Ventilation

___ • Aux Building Supply Fan(s)

___ • Aux Building Exhaust Fan(s)

___ d) Pull Inner Door closed and <u>HOLD</u> Inner Door in the CLOSED position until Locking Ring is engaged in the next step.

___ e) Press and hold the Inner Door CLOSE pushbutton until the Inner Door is fully closed.

___ f) Press the Outer Door OPEN pushbutton to fully open the Outer Door Locking Ring.

___ g) <u>WHEN</u> the Outer Door Locking Ring is full open, <u>THEN</u> close Outer Door and <u>HOLD</u> Outer Door in the CLOSED position until the Locking Ring is engaged in the next step.

___ h) Press and <u>HOLD</u> the Outer Door CLOSE pushbutton until the Outer Door is fully closed.

___ i) Notify Control Room of the status of Personnel Airlock doors.

- END -

13. 058AA2.02 088/NEW//H/3/3.3/3.6/6/

The crew is performing 1-ECA-0.0, Loss of ALL AC Power, and has NOT yet been able to restore Emergency Bus power.

Thirty (30) minutes have elapsed since the emergency busses were lost, and the crew is briefing an operator to perform 1-ECA-0.0, Attachment 2, Turbine Building Operations.

In order to ensure adequate 125V DC bus voltage, which ONE of the following identifies the actions required, **AND** includes the associated time frame per 1-ECA-0.0?

Within the next __________, stop the DC Turbine Oil Pump and the DC Air Side Seal Oil Pump.

A. 30 minutes, verify the turbine is stopped and generator hydrogen is vented, then

B. 60 minutes, verify the turbine is stopped and generator hydrogen is vented, then

C. 30 minutes, verify turbine speed is < 1000 rpm and generator hydrogen is vented, then

D. 60 minutes, verify turbine speed is < 1000 rpm and generator hydrogen is vented, then

a. Correct. DC Oil Pumps must be secured within one hour following loss of Emergency Bus power to the associated Battery Chargers. Prior to stopping these pumps the SRO must coordinate performance of the associated attachment, and verify completion of local actions within the specified time frame. Both items are critical to ensuring turbine does not rotate without lube oil (main lube oil pump pressure will be adequate at 1000 rpm, but insufficient at lower speeds) generator hydrogen does not vent via the seal which could result in a fire that would further conplicate recovery actions, and sufficent battery capacity remains available.

b. Incorrect. First part is incorrect, 30 minutes have already elapsed, an additional 60 minutes would jeopardize DC bus voltage; plausible since some actions are assumed to be initiated within certain time frames and the candidate who has only cursory knowledge may either assume this is the case or assume that the combined time is within 2 hrs and therefore find it acceptable while considering the other choice unreasonablly short. Second part is correct.

c. Incorrect. First part is correct as explained in answer a. Second part is incorrect but plausible since FCA-0 has similar actions and steps, breaking vacuum is delayed until speed is 1000 rpm based on concerns for blading damage, thus the candidate may default to this distractor, not realizing the subtle but important difference between the two postulated scenarios.

d. Incorrect. First part is incorrect but plausible as discussed in Distractor b. Second part is also incorrect but plausible as discussed in distractor c.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Loss of DC Power

Ability to determine and interpret the following as they apply to the Loss of DC Power:
(CFR: 43.5 / 45.13)

125V dc bus voltage, low/critical low, alarm

Tier: 1
Group: 1

Technical Reference: EOP ECA-0.0

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: The station does not have the specific alarm per se (only a battery charger trouble alarm). This item meets the intent of the KA since it solicits the candidates knowledge of both required action and bases for a situation where the battery busses do not have charger cabability and thus have limited capacity (time frame) that requires local operator actions as part of the mitigating strategy

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Operation of the DC Turbine Oil Pump and DC Air Side Seal Oil Pump for more than 1 hour with the Battery Chargers out of service may result in a degraded DC Bus voltage.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20. ___	CHECK DC BUS LOADS:	
	☐ a) Monitor DC Bus voltage	
	☐ b) Verify Turbine - STOPPED	☐ b) <u>WHEN</u> Turbine stops, <u>THEN</u> perform Step 20c. ☐ Continue with Step 20d.
	☐ c) Stop DC Turbine Oil Pump (bus 1-II)	
	☐ d) Verify Generator hydrogen - VENTED	☐ d) <u>WHEN</u> Generator hydrogen is vented, <u>THEN</u> perform Step 20e. ☐ Continue with Step 20f.
	☐ e) Stop DC Air Side Seal Oil Pump (bus 1-IV)	
	☐ f) Consult TSC or Plant Staff to determine other non-essential loads to be stopped	
*21. ___	CHECK ECST LEVEL - GREATER THAN 40%	☐ Make up from alternate sources using 1-AP-22.5, LOSS OF EMERGENCY CONDENSATE STORAGE TANK 1-CN-TK-1.

Where do we teach Design Bases reqts for SBO ?

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.____	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-I (1-EP-CB-12A):	
☐	a) Verify Generator Output Breaker G-12 - OPEN	☐ a) IF NOT open within 30 seconds of unit trip, THEN manually open G-12.
	b) As required, locally operate breakers on the following busses using 0-MOP-26.11, 4160-VOLT BREAKER LOCAL MANUAL OPERATION, or 0-MOP-26.10, 480-VOLT BREAKER LOCAL MANUAL OPERATION, because of loss of control power:	
☐	• 4160-Volt Bus 1C (1-EP-SW-03)	
☐	• 4160-Volt Bus 1H (1-EE-SW-01)	
☐	• 4160-Volt Bus 2G (2-EP-SW-04)	
☐	• 480-Volt Bus 1C1 (1-EP-SS-07)	
☐	• 480-Volt Bus 1C2 (1-EP-SS-04)	
☐	• 480-Volt Bus 1H (1-EE-SS-01)	
☐	• 480-Volt Bus 1H1 (1-EE-SS-03)	
☐	• 480-Volt Bus 1G2 (1-EP-SS-11)	
☐	• 480-Volt Bus 1G3 (1-EP-SS-12)	
☐	• 480-Volt Bus 2G2 (2-EP-SS-09)	
☐	• Exciter Field Breaker	
☐	• 15F1, 4160 Volt 1F Transfer Bus Feed 1H + 2J	

(STEP 6 CONTINUED ON NEXT PAGE)

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-II (1-EP-CB-12B):	
	☐ a) Check 1-RC-PCV-1455C, PRZR PORV - KEYSWITCH OFF	☐ a) Enter Action Statement of Technical Specification 3.4.12 because of loss of N_2 SOV.
	b) As required, locally operate breakers on the following busses using 0-MOP-26.11, 4160-VOLT BREAKER LOCAL MANUAL OPERATION, or 0-MOP-26.10, 480-VOLT BREAKER LOCAL MANUAL OPERATION, because of loss of control power:	
	☐ • 4160-Volt Bus 1B (1-EP-SW-02)	
	☐ • 480-Volt Bus 1B1 (1-EP-SS-05)	
	☐ • 480-Volt Bus 1B2 (1-EP-SS-08)	
	☐ • 480-Volt Bus 1B3 (1-EP-SS-09)	
	☐ • 15F3, 4160-Volt Trans Bus 1F Feed to Emer Bus 1H — 1H	
	☐ • 15E1, 4160 Volt 1E Transfer Bus Feed ← 2H bus	
7.___	CHECK DC BUS 1-II (1-EP-CB-12B) - CAN BE ENERGIZED	☐ Continue efforts to restore. RETURN TO Step 6.
8.___	RETURN TO STEP 1	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-III:	
☐	a) Locally check G-12 auxiliary switch position indicates - OPEN	☐ a) Verify G102-1 and G102-2 are open. IF NOT open, THEN send an Operator to 500KV Switchhouse to open G102-1 and G102-2.
	b) As required, locally operate breakers on following busses using 0-MOP-26.11, 4160-VOLT BREAKER LOCAL MANUAL OPERATION, or 0-MOP-26.10, 480-VOLT BREAKER LOCAL MANUAL OPERATION, because of loss of control power:	
☐	• 4160 Volt Bus 1A (1-EP-SW-01)	
☐	• 4160 Volt Bus 1J (1-EE-SW-02)	
☐	• 480 Volt Bus 1A1 (1-EP-SS-03)	
☐	• 480 Volt Bus 1A2 (1-EP-SS-06)	
☐	• 480 Volt Bus 1A3 (1-EP-SS-10)	
☐	• 480 Volt Bus 1J (1-EE-SS-02)	
☐	• 480 Volt Bus 1J1 (1-EE-SS-04)	
☐	c) Feed SGs with AFW because of loss of MFW Reg Valves and MFW Bypass Reg Valves	
☐	d) Check SG PORVs - CONTROLLING IN AUTO OR MANUAL	☐ d) Place SG PORVs in AUTO due to loss of Condenser Steam Dumps
☐	e) Check Unit 1 PCS - NOT FAILED	☐ e) Initiate 1-AP-42.1, LOSS OF UNIT 1 PLANT COMPUTER SYSTEM (PCS)

(STEP 6 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. ___	TAKE REQUIRED ACTIONS FOR LOSS OF DC BUS 1-IV (1-EP-CB-12D):	
	☐ a) Check 1-RC-PCV-1456, PRZR PORV - KEYSWITCH OFF	☐ a) Enter Action Statement of Technical Specification 3.4.12 because of loss of N_2 SOV.
	☐ b) As required, locally operate 15D3, 4160-Volt Trans Bus 1D Feed to Emer Bus 1J, using 0-MOP-26.11, 4160-VOLT BREAKER LOCAL MANUAL OPERATION, because of loss of control power	
	☐ c) Increase surveillance of Unit 1 Turbine Building and Transformer areas because of loss of Deluge System Alarms	
7. ___	CHECK DC BUS 1-IV (1-EP-CB-12D) - CAN BE ENERGIZED	☐ Continue efforts to restore. RETURN TO Step 6.
8. ___	RETURN TO STEP 1	

EMER TURB
OIL PP RUN
>15 MINUTES

NOTE: 1-TM-P-5 is powered from 125V DC Bus 1-II

1.0 Probable Cause

1.1 Emergency oil pump has been running for 15 minutes

2.0 Operator Action

2.1 Verify operation of 1-TM-P-5, Main Turbine Emergency Oil Pump 5.

2.2 Manually start 1-TM-P-1, Main Turbine AC Bearing Oil Pump, if available, as per 1-OP-41.1, Main Lube Oil System.

2.3 IF turbine is cold, THEN it may be removed from turning gear and emergency oil pump secured.

2.4 Notify SRO.

3.0 References

3.1 W Instruction Book 1250-C676 Vol. #1
3.2 11715-ESK-10BAK
3.3 11715-ESK-11A
3.4 11715-LO-008, NAPS Instrument Loop Book

4.0 Actuation

4.1 Time delay of 15, 30, 45, & 60 minutes initiated by 1-LO-PS-601
4.2 62-1TMGN02 (time delay closing relay)

AIR SIDE SO
BKUP PP RUN
>15 MINUTES

NOTE: 1-GM-P-8 is powered from 125V DC Bus 1-IV

1.0 Probable Cause

1.1 Air side Seal Oil pump failure or stopped
1.2 Failure of seal oil regulators or PDS8

2.0 Operator Action

2.1 Verify 1-GM-P-8, AIR SIDE SEAL OIL BACK-UP PUMP running.

2.2 Start 1-GM-P-2, SEAL OIL BACKUP PUMP as per 1-OP-42.1, Hydrogen Seal Oil System.

2.3 IF 1-GM-P-2 unavailable, THEN reduce H2 gas pressure to 2 psig as per 1-OP-43.1, Generator Gas System.

3.0 References

3.1 Seal oil diagram 637F207-1
3.1 11715-ESK-10BAJ

4.0 Actuation

4.1 Time delay of 15, 30, 45, & 60 minutes initiated by PDS 8
4.2 62-1GMSN10 (time delay closing relay)

Dominion

NORTH ANNA POWER STATION

FIRE CONTINGENCY ACTION

NUMBER	PROCEDURE TITLE	REVISION
0-FCA-0	FIRE PROTECTION - OPERATIONS RESPONSE (WITH EIGHT ATTACHMENTS)	12
		PAGE 1 of 8

PURPOSE

Provide immediate and long-term actions to take when a fire is reported at North Anna Power Station.

ENTRY CONDITIONS

- The Control Room has been notified of a fire at the station.
- The Control Room has been notified of the potential for a fire.
- Any fire alarm has sounded.

COMMON

CONTINUOUS USE

g) With Scene Leader concurrence, do the following to ventilate:

☐ • Start all Turbine Building exhaust fans

☐ • Open all Turbine Building outside intake louvers

☐ h) Close 2-GM-265, 2-GM-PDCV-219 Inlet Isolation Valve, located at the Generator Seal Oil unit.

* *

CAUTION: Securing the Seal Oil Pumps with the Turbine Rotating can cause Generator Seal damage. This is acceptable in the event of a Generator Seal Oil or Hydrogen fire.

* *

i) WHEN the Main Generator is vented, THEN stop the following Pumps:

☐ • 2-GM-P-1, AIR SIDE SEAL OIL PUMP

☐ • 2-GM-P-5, H2 SIDE SEAL OIL PUMP

* *

CAUTION: If the Condenser Vacuum Breaker is opened with Turbine speed greater than 1000 rpm, then blade damage may occur.

* *

☐ j) WHEN Turbine speed is less than 1000 rpm, THEN open 2-AS-MOV-200, CONDENSER VACUUM BREAKER.

☐ k) Close 2-MS-MOV-204, GLAND STEAM SUPPLY ISOL VALVE to secure Gland Steam to the turbine.

☐ l) IF excessive water from Fire fighting efforts is accumulating in the Turbine Building basement, THEN use the Turbine Building high capacity pumps for sump level control as directed by the SRO.

☐ m) Make-up to the Lube Oil Reservoir as required using 2-OP-41.1, Main Lube Oil System.

☐ n) Purge the Main Generator with $C0_2$ using the applicable section and steps for Purging Hydrogen from the Generator Using CO_2 in accordance with 2-OP-43.1, Operation Of The Generator Gas Systems.

g) With Scene Leader concurrence, do the following to ventilate:

☐ • Start all Turbine Building exhaust fans

☐ • Open all Turbine Building outside intake louvers

☐ h) Close 1-GM-265, 1-GM-PDCV-119 Inlet Isolation Valve, located at the Generator Seal Oil unit.

* *

CAUTION: Securing the Seal Oil Pumps with the Turbine Rotating can cause Generator Seal damage. This is acceptable in the event of a Generator Seal Oil or Hydrogen fire.

* *

i) WHEN the Main Generator is vented, THEN stop the following Pumps:

☐ • 1-GM-P-1, AIR SIDE SEAL OIL PUMP

☐ • 1-GM-P-5, H2 SIDE SEAL OIL PUMP

* *

CAUTION: If the Condenser Vacuum Breaker is opened with Turbine speed greater than 1000 rpm, then blade damage may occur.

* *

☐ j) WHEN Turbine speed is less than 1000 rpm, THEN open 1-AS-MOV-100, CONDENSER VACUUM BREAKER.

☐ k) Close 1-MS-MOV-104, GLAND STEAM SUPPLY ISOL VALVE to secure Gland Steam to the turbine.

☐ l) IF excessive water from Fire fighting efforts is accumulating in the Turbine Building basement, THEN use the Turbine Building high capacity pumps for sump level control as directed by the SRO.

☐ m) Make-up to the Lube Oil Reservoir as required using 1-OP-41.1, Main Lube Oil System.

☐ n) Purge the Main Generator with $C0_2$ using the applicable section and steps for Purging Hydrogen from the Generator Using CO_2 in accordance with 1-OP-43.1, Operation Of The Generator Gas Systems.

14. 064A2.06 089/BANK//L/3/2.9/3.3/6/

While attempting to synchronize and load the **1H EDG** per 1-PT-82H, 1H Emergency Diesel Generator Slow Start Test, the operator reported that EDG speed is oscillating and he is unable to sychronize the **1H EDG.**

The **1H EDG** is running with the output breaker open and speed is oscillating between 900 and 940 rpm.

Maintenance personnel at the EDG have requested to leave the EDG running to allow them to take some data prior to shutting it down.

In accordance with 1-PT-82H, the **1H EDG** should not be run unloaded for greater than ____ minutes **AND** in accordance with Technical Specifications a determination that **1J EDG** is NOT inoperable due to common cause failure is required to be completed within ______ hours.

A. 5 ; 4

B. 5 ; 24

C. 30 ; 4

D. 30 ; 24

a. Incorrect. First part is correct per the P&Ls of the PT. Second part is incorrect, but plausible since this is the action time for verification of opposite train equipment, and the candidate who lacks detailed knowledge ofthe TS may default to this distractor becuase they consider it more conservative.

b. Correct. First part correct as noted above. Second part is also correct per TS 3.8.1 condition b.

c. Incorrect. First part is incorrect but plausible since the candidate who lacks detailed system knowledge may consider the correct time frame unreasonably short and default to this distractor. Second part incorrect but plausible as discussed in Distractor a.

d. Incorrect. First part is incorrect but plausible as discussed in Distractor c. Second part is correct.

QUESTIONS REPORT

for NAPS 2010 SRO NRC Exam rev3

Emergency Diesel Generator (ED/G) System

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Operating unloaded, lightly loaded, and highly loaded time limit

Tier: 1
Group: 1

Technical Reference: 1-PT-82H & TS 3.8.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

4.0 PRECAUTIONS AND LIMITATIONS

______ 4.1 Comply with the following guidelines when marking steps N/A:

______ • IF the conditional requirements of a step do not require the action to be performed, THEN mark the step N/A.

______ • IF this test is being performed as a Partial PT or Post-Maintenance Test, THEN mark inappropriate steps N/A.

______ • IF any other step is marked N/A, THEN have the SRO approve the N/A and submit a Procedure Action Request (PAR).

______ 4.2 Do not perform testing on components undergoing maintenance.

______ 4.3 IF a step cannot be performed OR the required action is not achieved, THEN stop the test and notify the SRO.

______ 4.4 IF this test is being performed for surveillance, THEN record in the Action Statement Status Log any component NOT tested and the PT to be completed before returning the component to service.

______ 4.5 Record any component NOT tested and the reason on the Cover Sheet.

______ 4.6 Monitor EDG load carefully to make sure that the load does not exceed 3000 KW.

______ 4.7 No-Load Operation of the EDG should be minimized. The desired maximum limit is approximately 5 minutes of No-Load Operation at ≥ 900 RPM per occurrence. The time limit is intended as a conservative guideline for routine operations used to increase the life of the blower. If the EDG is run unloaded over 5 minutes at ≥ 900 RPM, the unloaded run time will be noted on the cover sheet and on 1-LOG-12, which is routed to the System Engineer.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Two emergency diesel generators (EDGs) capable of supplying the onsite Class 1E power distribution subsystem(s);

c. One qualified circuit between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System and one EDG capable of supplying the onsite Class 1E AC power distribution subsystem on the other unit for each required shared component; and

d. Required sequencing timing relays.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LCO 3.8.1.a offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit(s). <u>AND</u>	1 hour <u>AND</u> Once per 8 hours thereafter (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	A.3	Restore offsite circuit to OPERABLE status.	72 hours <u>AND</u> 17 days from discovery of failure to meet LCO
B. One LCO 3.8.1.b EDG inoperable.	B.1	Perform SR 3.8.1.1 for the required offsite circuits.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	B.2	Declare required feature(s) supported by the inoperable EDG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		(continued)

distractor

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE LCO 3.8.1.b EDG is not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE LCO 3.8.1.b EDG.	24 hours
	AND	
	B.4 Restore EDG to OPERABLE status.	14 days AND 17 days from discovery of failure to meet LCO
C. --------NOTE---------- Only applicable if Alternate AC (AAC) diesel generator (DG) or one or more EDG on the other unit is inoperable. ---------------------- One LCO 3.8.1.b EDG inoperable.	C.1.1 Restore inoperable AAC DG to OPERABLE status.	72 hours
	AND	
	C.1.2 Restore inoperable EDG(s) on other unit to OPERABLE status.	72 hours
	OR	
	C.2 Restore EDG to OPERABLE status.	72 hours

correct

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. --------NOTE---------- Separate Condition entry is allowed for each offsite circuit. --------------------- One or more required LCO 3.8.1.c offsite circuit(s) inoperable.	D.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND	
	D.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to a train concurrent with inoperability of redundant required feature(s)
	AND	
	D.3 Declare associated shared component inoperable.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One required LCO 3.8.1.c EDG inoperable.	E.1 Perform SR 3.8.1.1 for required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND	
	E.2 Declare required feature(s) supported by the inoperable EDG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	AND	
	E.3 Declare associated shared component inoperable.	14 days
F. --------NOTE---------- Only applicable if one or more LCO 3.8.1.b EDG(s) or AAC DG is inoperable. ---------------------- One required LCO 3.8.1.c EDG inoperable.	F.1.1 Restore inoperable AAC DG to OPERABLE status.	72 hours
	AND	
	F.1.2 Restore inoperable LCO 3.8.1.b EDG (s) to OPERABLE status.	72 hours
	OR	
	F.2 Declare associated shared component inoperable.	72 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two LCO 3.8.1.a offsite circuits inoperable.	G.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition G concurrent with inoperability of redundant required features
	<u>AND</u>	
	G.2 Restore one offsite circuit to OPERABLE status.	24 hours
H. One LCO 3.8.1.a offsite circuit inoperable. <u>AND</u> One LCO 3.8.1.b EDG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when Condition H is entered with no AC power source to any train. ----------------------------	
	H.1 Restore offsite circuit to OPERABLE status.	12 hours
	<u>OR</u>	
	H.2 Restore EDG to OPERABLE status.	12 hours
I. Two LCO 3.8.1.b EDGs inoperable.	I.1 Restore one EDG to OPERABLE status.	2 hours
J. Two required LCO 3.8.1.c EDGs inoperable.	J.1 Declare associated shared components inoperable.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. --------NOTE---------- Separate Condition entry is allowed for each sequencing timing relay. ---------------------- One or more required sequencing timing relay(s) inoperable.	K.1 Enter appropriate Conditions and Required Actions for any component made inoperable by inoperable sequencing timing relay(s).	Immediately
	AND	
	K.2.1 Place the component(s) with the inoperable sequencing timing relay in a condition where it cannot be automatically loaded to associated emergency electrical bus.	Immediately
	OR	
	K.2.2 Declare the associated EDG inoperable.	Immediately
L. Required Action and associated Completion Time of Condition A, B, C, G, H, or I not met.	L.1 Be in MODE 3.	6 hours
	AND	
	L.2 Be in MODE 5.	36 hours
M. Three or more of LCO 3.8.1.a and LCO 3.8.1.b AC sources inoperable.	M.1 Enter LCO 3.0.3.	Immediately

15. 065AA2.06 090/NEW//L/4/3.6/4.2/8/

Given the following conditions:

- Unit 1 is at 100% power.
- The team has entered 1-AP-28, Loss of Instrument Air.
- The crew has completed the immediate actions of 1-AP-28.

The OATC reports that Instrument Air pressure is 95 psig and lowering.

Which ONE of the following identifies the **MINIMUM** instrument air pressure allowed by 1-AP-28 before a trip is required **AND** includes the AFW flow required by 1-ES-0.1, Reactor Trip Response, following the trip (assuming RCPs are running)?

A. 70 psig ; 340 gpm

B. 70 psig ; 400 gpm

C. 75 psig ; 340 gpm

D. 75 psig ; 400 gpm

a. Incorrect. This is the minimum permitted by 1-Ap-28 and is a continuous action. Second part is incorrect but plausible since this is the minimum required for heat sink FRP, the candidate who lacks detailed knowledge of the procedure and bases may default to this distractor.

b. Correct. First part correct as noted above. Second part is also correct, ES-0.1 requires greater than the minimum value required just for heat sink, since as given RCPs are in operation.

c. Incorrect. First part is incorrect but plausible, 75 psig is a procedural value but it is used during leak isoaltion diagnostics. Second part incorrect but plausible as discussed in distractor a.

d. Incorrect. First part is incorrect as discussed in Distractor c. Second part is correct as discussed in answer b .

Loss of Instrument Air

Ability to determine and interpret the following as they apply to the Loss of Instrument Air:
(CFR: 43.5 / 45.13)

When to trip reactor if instrument air pressure is de-creasing

Tier: 1
Group: 1

Technical Reference: 1-AP-28, EOP ES-0.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

Dominion

NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-28	LOSS OF INSTRUMENT AIR (WITH SEVEN ATTACHMENTS)	30
		PAGE 1 of 25

PURPOSE

To provide instructions for recovering from a loss of Instrument Air.

ENTRY CONDITIONS

This procedure is entered when one of the following conditions exists:

- Rupture of Instrument Air System piping, or
- Failure of Instrument Air Compressors, or
- Transition from other plant procedure.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If a Reactor trip is required or occurs during the performance of this procedure, then this procedure should be used along with Emergency Operating Procedures to provide guidance for recovery from loss of Instrument Air.

[1]___ START AVAILABLE AIR COMPRESSORS:

- ☐ • 1-IA-C-1, Unit 1 IA Compressor
- ☐ • 1-SA-C-1, Unit 1 SA Compressor
- ☐ • 1-IA-C-2A, Unit 1 A Containment IA Compressor
- ☐ • 1-IA-C-2B, Unit 1 B Containment IA Compressor
- ☐ • 2-IA-C-1, Unit 2 IA Compressor
- ☐ • 2-SA-C-1, Unit 2 SA Compressor

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*2.___	CHECK INSTRUMENT AIR PRESSURE OUTSIDE CONTAINMENT:	answer
☐	a) Instrument Air pressure - LESS THAN 70 PSIG	☐ a) Monitor IA pressure. <u>IF</u> IA pressure decreases to less than 70 psig, <u>THEN</u> RETURN TO Step 2b. ☐ GO TO Step 3.
☐	b) Check Reactor - TRIPPED	☐ b) GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.
	c) Close Main Steam Trip Valves and Bypass Valves: ☐ • 1-MS-TV-101A ☐ • 1-MS-TV-101B ☐ • 1-MS-TV-101C ☐ • 1-MS-TV-113A ☐ • 1-MS-TV-113B ☐ • 1-MS-TV-113C	c) <u>IF</u> the Main Steam Trip Valves will not close, <u>THEN</u> close the Main Steam NRVs and NRV Bypass Valves: ☐ • 1-MS-NRV-101A ☐ • 1-MS-NRV-101B ☐ • 1-MS-NRV-101C ☐ • 1-MS-NRV-103A ☐ • 1-MS-NRV-103B ☐ • 1-MS-NRV-103C
☐	d) Verify RCS is <u>NOT</u> solid	d) Place the following equipment in PTL: ☐ • RCPs ☐ • Charging Pumps ☐ • PRZR Heaters
☐	e) GO TO Step 15	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.___	DETERMINE CAUSE FOR LOSS OF INSTRUMENT AIR: ☐ • Air compressor failure OR ☐ • Instrument Air Dryer failure OR ☐ • Instrument Air System piping rupture OR ☐ • Improper valve alignment OR ☐ • Foreign material in Instrument Air System	
4.___	CORRECT CAUSE FOR LOSS OF INSTRUMENT AIR	
5.___	VERIFY INSTRUMENT AIR PRESSURE - GREATER THAN 94 PSIG	☐ IF Instrument Air pressure is NOT trending to greater than 94 PSIG, THEN GO TO Step 15.
6.___	CHECK EITHER OF THE FOLLOWING INSTRUMENT AIR TRIP VALVES - CLOSED: ☐ • 1-IA-TV-102A ☐ • 1-IA-TV-102B	☐ GO TO Step 8.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: RCS pressure should be monitored due to possible PRZR Spray valve failure on Containment Instrument Air restoration.

* *

7.___ ESTABLISH CONTAINMENT INSTRUMENT AIR FROM OUTSIDE CONTAINMENT:

☐ a) Verify Containment Instrument Air pressure - GREATER THAN 75 PSIG

distractor

a) Do the following:

☐ 1) Maintain Outside Instrument Air pressure greater than 95 psig during Containment Instrument Air restoration.

2) Open both Instrument Air Trip Valves:

☐ • 1-IA-TV-102A

☐ • 1-IA-TV-102B

3) <u>IF</u> Outside Instrument Air pressure decreases to 95 psig, <u>THEN</u> do the following:

a. Close at least one of the following valves:

☐ • 1-IA-TV-102A

☐ • 1-IA-TV-102B

☐ b. <u>WHEN</u> Outside Instrument Air pressure stabilizes, <u>THEN</u> RETURN TO Step 7.

☐ 4) GO TO Step 8

(STEP 7 CONTINUED ON NEXT PAGE)

Dominion

NORTH ANNA POWER STATION

EMERGENCY PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-ES-0.1	REACTOR TRIP RESPONSE (WITH THREE ATTACHMENTS)	28
		PAGE 1 of 21

PURPOSE

To provide instructions to stabilize and control the plant following a Reactor Trip without a Safety Injection.

TIME CRITICAL ACTIONS

ENTRY CONDITIONS

This procedure is entered from:

- 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1. ___	CHECK RCS AVERAGE TEMPERATURE:	
	a) Check Temperature Control:	a) Do the following:
	• STEAM DUMPS - CONTROLLING:	IF temperature is less than control value AND decreasing, THEN:
☐	• STABLE AT 547°F	☐ 1) Stop dumping steam.
	OR	☐ 2) Verify SG Blowdown Trip Valves are closed.
☐	• TRENDING TO 547°F	☐ IF NOT, THEN manually close valves.
	OR	
	• SG PORVs - CONTROLLING:	☐ 3) Adjust total AFW flow to maintain greater than 400 gpm (340 gpm with RCPs OFF) until at least one SG narrow range level is greater than 11%.
☐	• STABLE AT 551°F	
	OR	
☐	• TRENDING TO 551°F	
		4) IF cooldown continues, THEN close the following:
		☐ • MSTVs
		☐ • MSTV Bypass Valves
		☐ 5) GO TO Step 2.

(STEP 1 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* 1. CHECK RCS AVERAGE TEMPERATURE: (Continued)

IF temperature is greater than control value AND increasing, THEN do the following:

- ☐ • Dump steam to the Condenser

  OR

- ☐ • Dump steam using SG PORVs

  OR

- • Dump steam using Decay Heat Release Valve:

  1) Locally open isolation valve(s) for NON-RUPTURED SG(s) to Decay Heat Release Valve:

     - ☐ • 1-MS-19, A Steam Line to 1-MS-HCV-104 Non-Return Valve
     - ☐ • 1-MS-58, B Steam Line to 1-MS-HCV-104 Non-Return Valve
     - ☐ • 1-MS-96, C Steam Line to 1-MS-HCV-104 Non-Return Valve

  ☐ 2) Locally open 1-MS-20, Decay Heat Release Valve Upstream Isolation Valve.

  ☐ 3) Manually open 1-MS-HCV-104, Decay Heat Release Valve.

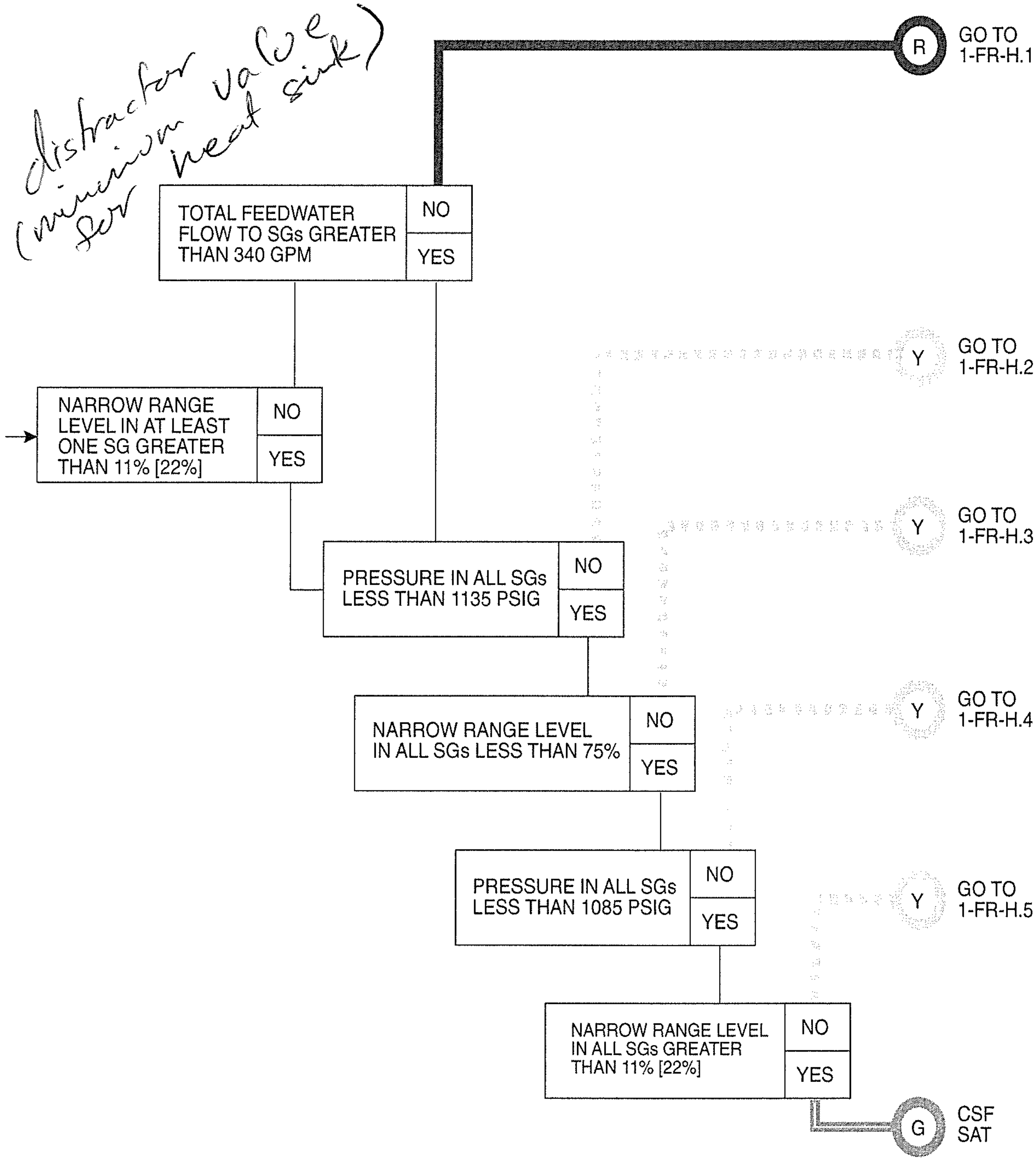
☐ GO TO Step 1.b.

answer

☐ b) Adjust total AFW flow between SGs to maintain greater than 400 gpm (340 gpm with RCPs OFF) until at least one SG narrow range level is greater than 11%. distractor

NUMBER 1-F-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 7	HEAT SINK	PAGE 1 of 1

distractor value (minimum for heat sink)



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16. 068AG2.2.40 091/NEW//L/3/3.4/4.7/8/

Which ONE of the following identifies the component whose isolation switch is required by TR 7.5, Appendix R Alternate Shutdown Equipment, **AND** includes the **required action and completion time** if the minimum requirement is not met?

A. 1-CH-TV-1204A, Letdown Isol Valve;
Implement hourly fire watch within 14 days.

B. 1-CH-TV-1204A, Letdown Isol Valve;
Restore equipment to functional status within 14 days or be in Mode 3 within the next 6 hours.

C. 1-RC-LCV-1460A, Letdown Isol Valve;
Implement hourly fire watch within 14 days.

D. 1-RC-LCV-1460A, Letdown Isol Valve;
Restore equipment to functional status within 14 days or be in Mode 3 within the next 6 hours.

a. Incorrect. Plausible since candidate who has only cursory knowledge of App. R requirements (isolating the RCS) may default to this distractor since the subject valve is a containment isolation valve. Second part is correct, but fire watch requirements vary and the candidate who lacks detailed knowledge of the TRM may eliminate this choice solely because they feel the time frame is excessive (i.e. why not a requirement to implement upon discovery?).

b. Incorrect. First part incorrect but plausible as noted above. Second part is incorrect, the TRM allows 60 days to restore it to a functional status, and again plausible since the candidate may erroneously conclude that if the function could not be restored the action is warranted.

c. Incorrect. First part is correct as stated in TR 7.5, table 7.5-1. Second part also correct as discussed in Distractor a.

d. Correct. First part is correct as discussed in Distractor c. Second part is incorrect as noted in distractor b.

Control Room Evacuation

Ability to apply Technical Specifications for a system.
(CFR: 41.10 / 43.2 / 43.5 / 45.3)

Tier: 1
Group: 2

Technical Reference: TR 7.5 and TR table 7.5-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

7.0 FIRE PROTECTION

7.5 Appendix R Alternate Shutdown Equipment

TR 7.5 The equipment listed in Table 7.5-1 shall be FUNCTIONAL.

APPLICABILITY: At all times.

see pg 7.5-3

ACTIONS

NOTES

1. TR 3.0.3 is not applicable.
2. Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more components listed in Table 7.5-1 nonfunctional.	A.1 Implement hourly fire watch in the area(s) associated with the nonfunctional equipment identified in Table 7.5-1.	14 days answer
	AND	
	A.2 Restore equipment to FUNCTIONAL status.	60 days
B. Required Action A.2 not completed within the specified Completion Time.	B.1 Submit a Plant Issue.	Immediately

TRM SURVEILLANCE REQUIREMENTS

None

see next pg

Table 7.5-1 (page 1 of 3)
Appendix R Alternate Shutdown Equipment

EQUIPMENT	FIRE WATCH LOCATION
CROSS-CONNECT VALVES BETWEEN UNITS: Charging Pump Cross-Connect: 1-CH-550,2-CH-408 Component Cooling Water (CCW) Cross-Connect: 1-CC-49	Auxiliary Building (AB) AND Both unit's Emergency Switchgear Room (ESGR) & Cable Vault and Tunnel (CV&T)
Charging pumps: maintain at least one FUNCTIONAL pump available on the shutdown unit when opposite unit is running.	AB AND Operating unit's ESGR & CV&T
Auxiliary Service Water Pumps (ASWPs)[a] and associated valves: 1-SW-P-4 2-SW-P-4	Service Water Pumphouse
Appendix R Distribution Panels 1(2)-EP-CB-002	ESGR AND CV&T (Unit w/FUNCTIONAL panel)
DIESEL GENERATOR LOCAL CONTROL PANELS[b]: Diesel Generator 1H Diesel Generator 2H	N.A.
Transfer and local control switches on MCC's for residual heat removal (RHR) pumps Transfer and local control switches on MCC's for CCW pumps Transfer and local control switches on MCC's for service water pumps	N.A.

(a) ASWPs are required for 10 CFR 50 Appendix R, Section III.G.2 only.

(b) Not required when associated emergency diesel generator is nonfunctional.

Table 7.5-1 (page 2 of 3)
Appendix R Alternate Shutdown Equipment

answer distractor see analysis for discussion for plausibility of alternatives

EQUIPMENT	FIRE WATCH LOCATION
Isolation switches for closing the following Unit 1 valves: 1-RC-PCV-1456 and 1-RC-PCV-1455C 1-CH-LCV-1460A 1-CH-HCV-1137 1-RC-SOV-101A,101B,102A, and 102B	Unit 1 ESGR <u>AND</u> CV&T
Isolation switches for closing the following Unit 2 valves: 2-RC-PCV-2456 and 2-RC-PCV-2455C 2-CH-LCV-2460A 2-CH-HCV-2137 2-RC-SOV-201A,201B,202A, and 202B	Unit 2 ESGR <u>AND</u> CV&T
The following ventilation equipment: Isolation and transfer switches: 1-HV-AC-6 & 2-HV-AC-7, Chillers: 1-HV-E-4B, 1-HV-E-4C, 2-HV-E-4B & 2-HV-E-4C and their associated chilled water pump, condenser water pumps and motor operated valves and controls	Opposite unit's ESGR <u>AND</u> Chiller Room
DEDICATED VENTILATION SYSTEMS 1-HV-F-74 1-HV-F-75A 1-HV-F-75B	AB
Isolation switches in the CV&T for steam generator power operated relief valves	Same unit's ESGR
MAIN STEAM TRIP VALVE DEDICATED SHUTDOWN SYSTEM 1-MS-SOV-101A-6 & A-7 1-MS-SOV-101B-6 & B-7 1-MS-SOV-101C-6 & C-7 2-MS-SOV-201A-6 & A-7 2-MS-SOV-201B-6 & B-7 2-MS-SOV-201C-6 & C-7	Same unit's ESGR <u>AND</u> CV&T

Table 7.5-1 (page 3 of 3)
Appendix R Alternate Shutdown Equipment

EQUIPMENT	FIRE WATCH LOCATION
Equipment that is required for the RHR pump repair procedure 0-ECM-0204-01[c]	Both units' CV&T AND ESGR
Equipment in the Appendix R locker	N.A.
Refueling water storage tank (RWST) Minimum 51,000 gallons is required in tank during all operating modes, if opposite unit is operating. The flow path from the RWST to an available charging pump on the shutdown unit is required when the opposite unit is operating.	AB AND Opposite unit's ESGR and CV&T

(c) Majority of materials and equipment are contained within the Appendix R storage area within Warehouse 6. Additional required equipment is contained in the Electric Shop or Unit 2 Normal Switchgear Room cage area or Tool Crib. See procedure 0-EPM-2304-02 for a list of materials and equipment required.

17. G2.1.31 092/NEW//L/3/4.6/4.3//

Unit 1 experienced a loss of offsite power followed by a small-break LOCA.

Operators are implementing 1-ES-1.2, Post-LOCA Cooldown and Depressurization.

The following conditions exist:

- Containment pressure has been steadily increasing and is now 21 psia and slowly increasing.
- Containment radiation has been steadily increasing and is now 3.5R/hr and stable.

Assuming these plant conditions remain unchanged, which ONE of the following identifies how the CRDM Fans are operated by 1-ES-1.2, **AND** includes the reason?

A. CRDM fans are left running to provide additional containment cooling.

B. CRDM fans are stopped to reduce emergency bus loading.

C. CRDM fans are stopped to prevent sump blockage due to degradation of the aluminum fan blading.

D. CRDM fans are stopped because of motor overload concerns.

a. Incorrect but plausible since the candidate who lacks detailed knowledge of this recent change to the EOPs would conclude that this choice would be beneficial and desirable since removing heat from containment will help control containment pressure.

b. Incorrect but plausible since again, the candidate who lacks detailed knowledge of this recent change to the EOPs may erroneously conclude that concerns for bus loading justify shutting down these components since they are not needed for accident mitigation.

c. Correct. As already noted this is a recent (site specific) EOP change; based on analysis that concluded there is a negative impact from interaction of the containment environment with the aluminum fan blades of the CRDM fans these fans are stopped to avoid any negative consequence to recirc sump performance. Again the candidate who relys on past knowledge would likely discount this choice since there is no clear correlation between the sump and the fans.

d. Incorrect but plausible since the candidate who lacks detailed knowledge of this recent change to the EOPs could easily rationalize that since there is a valid concern for motor overload, (considering the containment conditions), the fans would be stopped to avoid potentially jeopardizing an Emergency Bus.

QUESTIONS REPORT

Conduct of operations

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.
(CFR: 41.10 / 45.12)

Tier: 3

Technical Reference: 1-ES-1.2 & background document (note: Plant specific issue).

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

Dominion

NORTH ANNA POWER STATION

EMERGENCY PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-ES-1.2	POST-LOCA COOLDOWN AND DEPRESSURIZATION (WITH FIVE ATTACHMENTS)	20 PAGE 1 of 29

PURPOSE

To provide instructions to cool down and depressurize the RCS to Cold Shutdown conditions following a loss of Reactor Coolant inventory.

ENTRY CONDITIONS

This procedure is entered from:

- 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, or
- 1-ES-1.1, SI TERMINATION.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	RESET BOTH TRAINS OF SI	☐ Perform 1-AP-0, RESETTING SI LOCALLY, while continuing with this procedure.
2.___	RESET CONTAINMENT ISOLATION SIGNALS: ☐ a) Reset both Trains of Phase A Isolation ☐ b) Reset both Trains of Phase B Isolation, if actuated	
3.___	VERIFY OUTSIDE INSTRUMENT AIR SUPPLYING CONTAINMENT	Do the following: ☐ a) Start at least one Air Compressor. b) Manually open Containment Instrument Air Trip Valves: ☐ • 1-IA-TV-102A ☐ • 1-IA-TV-102B
*4.___	VERIFY ALL AC BUSSES - ENERGIZED BY OFFSITE POWER	☐ Initiate 0-AP-10, LOSS OF ELECTRICAL POWER, to restore offsite power. <u>IF</u> required, <u>THEN</u> manually load the following equipment upon restoration of any AC Emergency Bus: ☐ • CRDM Fans ☐ • Containment Air Recirc Fans ☐ • PRZR Heaters

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: PRZR heaters should not be energized until directed by this procedure to ensure heaters are covered.

* *

5.____ PUT ALL PRZR HEATER CONTROLS IN PTL

*6.____ CHECK IF QUENCH SPRAY IS REQUIRED:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a) Verify ALL of the following:	☐ a) GO TO Step 7.
☐ • Quench Spray Pumps - NOT RUNNING	
☐ • Containment pressure - HAS EXCEEDED 20 PSIA	
☐ • Containment pressure - GREATER THAN 13 PSIA	
☐ • Containment Radiation - GREATER THAN 2 R/hr	
☐ b) Manually actuate CDA	
☐ c) Verify CC Pumps - TRIPPED	☐ c) Stop CC Pumps.
☐ d) Stop all RCPs	
☐ e) Verify QS Pumps - RUNNING	☐ e) Manually start QS Pumps.
f) Verify QS Pump Discharge MOVs - OPEN:	☐ f) Manually open valves.
☐ • 1-QS-MOV-101A	
☐ • 1-QS-MOV-101B	

(STEP 6 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6.	CHECK IF QUENCH SPRAY IS REQUIRED: (Continued)	answer, see background (step deviation) document for completion of answer & additional info.
	g) Stop ALL CRDM fans:	
☐	• 1-HV-F-37A and 1-HV-F-37D	
☐	• 1-HV-F-37B and 1-HV-F-37E	
☐	• 1-HV-F-37C and 1-HV-F-37F	
☐	h) On the Unit 1 Ventilation Panel, verify 1-SW-TV-101A&B SERVICE WATER SUPPLY & RETURN TO RECIRC AIR FANS - SWITCH IN CLOSE POSITION	☐ h) Place switch in CLOSE.
☐	i) Initiate 1-E-0, ATTACHMENT 2, VERIFICATION OF PHASE B ISOLATION	
☐	j) Initiate 1-E-0, ATTACHMENT 3, PRIMARY PLANT VENTILATION ALIGNMENT	

EOP Step	ERG Step	Sequence No.
5	5	(1)

Changed the wording from "OFF" to "PTL".
To ensure that the heaters do not come on they are placed in PTL.

6 ----

Added step to check if quench spray is required.

For primary break LOCAs that do not cause containment pressure to exceed <M.27> it has been shown that no fuel damage is expected, so that containment iodine is a not a concern. For larger breaks, it is prudent to initiate containment spray for dose reduction and to ensure iodine retention in the containment sump. Reference CTS 02-94-1204 item 001, Operating Experience Review PT21 93-24-S1, Westinghouse NSAL-93-016, and NSA memo # 93-172.

Check if CDA is required and initiate CDA when Containment pressure exceeds <M.27> and Containment Radiation is greater than <M.6>. Containment Radiation is addressed vice the requirement of all SG pressures stable or under operator control. The containment pressure & radiation conditions will be indicative of fuel failure and address the faulted SG. If a small break LOCA occurs and the CDA setpoint is not reached, then a Phase B Containment isolation is desired. The CDA will provide Phase B isolation and Casing Cooling start will provide additional NPSH for recirculation mode. (Reference: EOP Basis Document Change 2010-001)

Deleted CAT Tank level as an indication that sufficient NaOH had been added for Iodine retention. Deleted opening CAT valves. Tank level indication not required since CDA is required for Phase B isolation and ES-1.3 will ensure sufficient NaOH addition. Opening CAT valves not required since CDA and QS start will open valves. (Reference: EOP Basis Document Change 2010-001)

7C 6C

Caution on restarting LHSI pumps was reworded.
The first sentence on monitoring RCS pressure was deleted because the statement is implicit in the caution (not needed). IF, THEN logic was used in the caution to make it clearer to the operator.

Dominion

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THIS DOCUMENT SHALL BE VERIFIED
TO A CONTROLLED SOURCE AS
REQUIRED TO PERFORM WORK

DATE: 02/04/09

TO: RECORDS

FROM: JULIUS COPPA

[redacted]

SUBJECT: **CHANGE TO EOP BASIS DOCUMENT NUMBER**
2008-09 for FR-S.l & E-3 and
2008-10 for E-1, ECA-0.2, ECA-3.1, ECA-3.3, ES-1.2, FR-H.l, and FR-Z.4

Included with this memo are two EOP Basis Document Changes (EOP Step Difference Evaluations - SDEs) for subject SDEs. A copy of these changes requires distribution to holders of the Integrated Documents Sets.

JULIUS COPPA

Operations Procedure Writer
NAPS Procedures—TSB

To: North Anna PROCEDURES
Attn: J. Coppa

December 29, 2008

From: D. L. Horn

Nuclear Analysis and Fuel

Change to EOP Basis Document Number: 2008-10
Keywords: NAPS1 NAPS2 EOP SDE

This EOP Basis Document Change addresses changes to E-1, ECA-0.2, ECA-3.1, ECA-3.3, ES-1.2, FR-H.1, and FR-Z.4 which ensure the CRDM fans are secured during Containment Spray to minimize aluminum transport to the Containment Sump Strainers.

All documents added a step to stop all CRDM fans after the Quench Spray system is initiated. Listed below are the procedure numbers and the step numbers which have added this requirement.

EOP Number	Step Number Accomplishing Requirement
1-E-1 and 2-E-1	7.c
1-ECA-0.2 and 2-ECA-0.2	10.c
1-ECA-3.1 and 2-ECA-3.1	6.c
1-ECA-3.3 and 2-ECA-3.3	19.c
1-ES-1.2 and 2-ES-1.2	6.c
1-FR-H.1 and 2-FR-H.1	Att. 5, 8.c
1-FR-Z.4 and 2-FR-Z.4	10 and 16

Additionally, the step difference evaluation (SDE) document for FR-Z.4 has been revised to account for procedure flow changes to ensure step numbering consistency. The changes are minor adjustments consistent with the intent of the emergency procedure.

The change incorporates CA082498. There is no impact on EOP Setpoints.

Engineering Programs/Maint. Rule Effect: None

Implementation Schedule: ASAP

Attached to this memo are the revised NAPS EOP step difference evaluation documents for E-1, ECA-0.2, ECA-3.1, ECA-3.3, ES-1.2, FR-H.1, and FR-Z.4. If you have any questions regarding this issue, please call.

C L Tiernan for DL Horn per telecon
D. L. Horn (no copy)

Prepared By: C L Tiernan
C. L. Tiernan

Reviewed By: Russ C. Anderson
R. C. Anderson

Attachments: SDE revisions dated 12/29/2008 for E-1, ECA-0.2, ECA-3.1, ECA-3.3, ES-1.2, FR-H.1, and FR-Z.4
cc: S.H.Shen—IN3NW
NSA File

18. G2.1.38 093/NEW//L/3/3.7/3.8//

Given the following conditions:

- Unit 1 was initially at 100% power.
- The crew is performing 1-E-0, Reactor Trip or Safety Injection, in response to a spurious SI.
- The Unit Supervisor (US) is reading 1-E-0 and directs the RO to reset both trains of SI.

Which ONE of the following describes how the RO will carry out this direction?

A. Repeat back the US's order, obtain the required peer check from the BOP, proceed with the manipulation unless the SRO states "incorrect".

B. Repeat back the US's order, proceed with the manipulation unless the SRO states "incorrect", a peer check may be obtained but is NOT required.

C. Repeat back the US's order, obtain the required peer check from the BOP, ensure a repeat back from the US is received prior to performing the manipulation.

D. Repeat back the US's order, ensure an acknowledgement is obtained from the US prior to performing the manipulation, a peer check may be obtained but is NOT required.

a. Incorrect. Plausible since most would feel a peer check is appropriate, however is is not expressly requried. Second part is incorrect only because a manipulation is involved.

b. Incorrect. Again because of a manipulation involved the repeat back is required prior to taking action, the second part is true, and while the need to do a peer check for this may be debatable the fact that it is not required but may be obtained if desired by the operator is true.

c. Incorrect. As noted above there is no expressed requriement for a peer check.

d. Correct. As previously discussed both parts are true, a repeat back is procedurally required, and while the peer check is not explicitly required for the task the operator may request one if they feel it is prudent.

Conduct of operations

Knowledge of the station's requirements for verbal communications when implementing procedures. (CFR: 41.10 / 45.13)

Tier: 3

Technical Reference: PI-AA-5000, OP-AP-104

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

ATTACHMENT 8

(Page 1 of 5)

Human Performance Event Free Tool - Clear Communication (Fundamental Tool)

Purpose

The goal of clear communication is mutual understanding between two or more people, especially communication involving technical information related to plant operation or personnel safety. Clear communication is likely the most important defense in the prevention of errors and events. The verbal exchange of information, especially face-to-face, is the most frequent form of communication, an important team behavior. However, verbal communication possesses a greater risk of misunderstanding compared to written forms of communication. Misunderstandings are most likely to occur when the individuals involved have different understandings, or mental models, of the current work situation or use terms that are potentially confusing. Therefore, confirmation of verbal exchanges of operational information between persons must occur to promote understanding and reliability of the communication. **[Reference 5.4.18]**

The Clear Communication tool contains three subsets of information, three-way communication, use of the phonetic alphabet, and turnover. Three-way communication requires three verbal exchanges between a sender and a receiver to promote a reliable transfer of information and understanding. It is used in communication of changes to physical plant equipment during work activities via face-to-face, telephone, or radio. If not providing direction or important plant information, other work-related verbal communication may be conducted less formally, using two-way communication, for example (sender speaks, receiver acknowledges he/she heard/understood message).

Three-Way Communication

Background: The person originating the communication is the sender and is responsible for verifying that the receiver understands the message as intended. The receiver makes sure he or she understands what the sender is saying. First, the sender gets the attention of the receiver and clearly states the message. Second, the receiver repeats the message. Paraphrasing is allowed, which helps the sender know if the receiver understands the message. During this exchange, the receiver restates equipment-related information exactly as spoken by the sender. Third, the sender informs the receiver whether the message is properly understood, or corrects the receiver and restates the message.

The weakest link of a communication is often the third leg, because the sender may assume the receiver heard the message. If the receiver does not understand the message, he or she should ask for clarification, confirmation, or repetition of the message. If practical, it is helpful to support three-way communication with other information aids, such as procedures, work packages, and indicators.

When to Use the Tool [Reference 5.4.19]

Consider using three-way communication in verbal conversations involving:

- Operation or alteration of plant equipment
- Condition of plant equipment or the value of an important parameter
- Performance of steps or actions using an approved procedure
- Task assignments that impact plant equipment or plant activities
- Safety of personnel, the environment, or the plant

ATTACHMENT 8

(Page 2 of 5)

Human Performance Event Free Tool - Clear Communication (Fundamental Tool)

How to Use the Tool

a. Sender states the message.

- If practical, the sender positions himself or herself in front of the intended receiver (preferably face to face).
- The sender gets the attention of the receiver, such as using first names.
- Sender states the message clearly and concisely.

b. Receiver acknowledges the sender.

- The receiver repeats the message. Paraphrasing is allowed.
- Equipment designators and nomenclature as stated by the sender are repeated word for word.
- The receiver asks questions to verify his or her understanding of the message.

c. Sender acknowledges the receiver's reply.

- If the receiver understands the message, then the sender responds with "Correct."
- If the receiver does not understand the message, the sender responds with "Wrong" and restates the original message

d. If corrected, receiver acknowledges the corrected message, again paraphrasing the message in his or her own words.

<u>At-Risk Practices to Avoid</u>:

- Sender not using receiver's name to get receiver's attention
- Sender speaking from behind the receiver or not making eye contact when it is practical to do so
- Sender or receiver not stating his or her name and work location when using a telephone or radio
- Sender attempting to communicate with someone already engaged in another conversation
- Sender stating too much information or multiple actions in one message
- Sender not giving enough information the receiver needs to understand the message
- Sender not verifying receiver understood the message
- Receiver reluctant to ask for clarification of the message
- Receiver taking action before the communication is complete
- Receiver not writing the message down if there are more than two items to remember
- Receiver given information unrelated to the immediate task
- Receiver mentally preoccupied with another task
- Overusing the tool for non-operational communications
- Not using three-way communication in order to expedite the task
- Message not being stated loudly enough to be heard
- Enunciating words poorly

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Nuclear Fleet

Administrative Procedure

Title: Emergency and Abnormal Operating Procedures

Procedure Number	Revision Number	Effective Date and Approvals On File
OP-AP-104	2	

Revision Summary

Revised to clarify the reading of NOTEs and CAUTIONs:

- Updated 3.4.1.e (EOPs) as follows:
  - OLD - "During performance of all EOPs, the US shall read aloud all applicable NOTEs and CAUTIONs in their entirety the first time they are encountered. When encountered again, the US may read or paraphrase the applicable NOTEs and CAUTIONs. The ROs will paraphrase the Caution or Note and must not respond: understand the NOTEs and CAUTIONs, as the paraphrase. The US will complete the 3-way communication. If the NOTE or CAUTION pertains only to the US or is not applicable at the time, the US is not required to read aloud."
  - NEW - "During performance of all EOPs, the US shall read aloud all applicable NOTEs and CAUTIONs in their entirety the first time they are encountered. When encountered again, the US may read or paraphrase the applicable NOTEs and CAUTIONs. The ROs will paraphrase or acknowledge the NOTE or CAUTION. If the NOTE or CAUTION pertains only to the US or is not applicable at the time, the US is not required to read aloud."
- Updated 3.7.2 (fourth bullet) (AOPs) as follows:
  - OLD - "A NOTE or CAUTION may be directed to applicable crew members; in such cases, repeat-back of the paraphrased NOTE or CAUTION is expected"
  - NEW - "A NOTE or CAUTION may be directed to applicable crew members; in such cases, the applicable crew member will paraphrase or acknowledge the NOTE or CAUTION"

Functional Area Manager: **Manager Nuclear Operations**

INFORMATION USE

1.0 PURPOSE

This procedure establishes format and use requirements for transient response procedures used to respond to abnormal and emergency situations.

2.0 SCOPE

This procedure provides guidance to personnel who develop and use emergency and abnormal operating procedures. The requirements apply at all times when involved in plant operations. Additional guidance is provided in documentation that is plant or site specific.

3.0 INSTRUCTIONS

3.1 Transient Response

Licensed Operator

3.1.1 **COMMIT** the immediate actions of emergency and abnormal operating procedures to memory.

> **NOTE:** If a plant transient occurs that could challenge safety, the Shift Manager or Unit Supervisor may direct operators to take actions without first reviewing a procedure.

3.1.2 After taking actions during transient that could challenge safety, **REVIEW** the appropriate procedure to ensure that all actions were performed correctly.

3.1.3 **REPORT** any component that requires manual control to the EOP Implementer when the applicable step is read.

3.1.4 **ENHANCE** performance by the following techniques:

a. Use 3-way communications when transferring information that may result in changes to plant configuration. At other times, streamline verbal communications by being concise and to the point.

b. Conduct transient and focus briefings of short duration to optimize crew performance. Conduct a transient briefing following completion of EOP immediate actions, upon transition to a new emergency procedure, or when requested by a team member.

c. Use peer checking judiciously during transient response. Peer checking is inappropriate when it involves delaying time-critical actions or distracting operators during key actions. Reserve peer checking for actions of a critical nature such as those that are irreversible.

d. Understand the mitigative strategies and bases for emergency and abnormal operating procedures.

e. Use placekeeping.

19. G2.2.20 094/NEW//L/3/2.6/3.8//

Given the following conditions:

- Unit 1 is at 100% power.
- Maintenance is preparing to troubleshoot an indicating light bulb issue with 1-QS-P-1A.
- 1-QS-P-1A will need to be tagged out to support the troubleshooting.
- An SRO is reviewing MA-AA-103, Attachment 2, Troubleshooting Sheet.

IAW the Troubleshooting Sheet, the SRO must verify ________________ operable.

A. ONLY the 1H EDG

B. BOTH 1H EDG and 1-QS-P-1B

C. ONLY the 1J EDG

D. BOTH 1J EDG and 1-QS-P-1B

a. Incorrect. Plausible since the candidate who does not have detailed knowledge of the procedure may assume that this would be desirable based on a potential for relaying action to occur while working around the affected bus. Given this erroneous assumption it would be logical that one would want the Emergency power supply to the bus available in the event relaying tripped the normal supply.

b. Incorrect. First item incorrect but plausible as noted above; second part is correct.

c. Incorrect. Plausible since there have been issues related to opposite Train EDGs in the past and the candidate who has only cursory knowledge of the procedure and resent change to it, may conclude that this is the only check made and other requirements would be contained elsewhere.

d. Correct. As discussed in Distractor c the procedure was resently changed for a check of the opposite train emergency power supply. The opposite Train equipment is also checked, but again the candidate who lacks detailed knowledge of the troubleshooting procedure may conclude that since this check is incorporated elsewhere, it would not be necessary to include it here and thus discount this choice.

Equipment Control

Knowledge of the process for managing troubleshooting activities.
(CFR: 41.10 / 43.5 / 45.13)

Tier: 3

Technical Reference: MA-AA-103

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

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Administrative Procedure

Title: Conduct of Troubleshooting

Document Number	Revision Number	Effective Date and Approvals On File
MA-AA-103	4	

Revision Summary

Revised to:

- Update Plant Manager responsibilities to add assigning the TTL for high risk troubleshooting.
- Clarify requirements for completion of activity checklist / pre-screening (i.e., in accordance with VPAP-3001 or GNP-0.4.04.01) during troubleshooting plan development.
- Update form Troubleshooting Sheet (Attachment 2) - 730600(Dec 2009) to ensure appropriate personnel are listed and add additional questions.

Details:

- Moved revised 3.3.4 which replaced 3.7.9 to ensure at least a preliminary plan has been developed prior to determining risk and rigor.
- Revised 3.7.1 to clarify requirements for completion of activity checklist / pre-screening (i.e., in accordance with VPAP-3001 or GNP-0.4.04.01) during troubleshooting plan development.
- Added 5.2.1.b to update Plant Manager responsibilities:
  - Added: Assigning the TTL for high risk troubleshooting.
- Revised 5.2.2.b to update Manager Nuclear Maintenance responsibilities for consistency with added 5.2.1.b:
  - Old: Assigning the TTL for high and medium risk troubleshooting.
  - New: Assigning the TTL for medium risk troubleshooting.
- Updated definitions:
  - Revise 5.3.3, Non-Intrusive Troubleshooting, to add “circuit test points” to the Non-Intrusive Troubleshooting examples.
  - Revised 5.3.5, Technical lead (TL), for consistency with 3.4.3 note.
  - Revised 5.3.7, Troubleshooting Team Lead (TTL), for consistency with revised responsibilities.
- Deleted reference “Dominion Cause Evaluation Handbook” which is being replaced by revised fleet procedure PI-AA-300 and three related new Guidance and Reference Documents.
- Revised form Troubleshooting Sheet (Attachment 2) - 730600(Dec 2009) to ensure appropriate personnel are listed and add additional questions:
  - Old: Name of Personnel Having Knowledge of the Problem (Mark N/A if Simple Troubleshooting).
  - New: Name of Personnel Having Knowledge of the Problem.
  - Old: First Line Supervisor Approval, if required.
  - New: Troubleshooting Team Lead (TTL) Approval, if required.
  - Added Yes / No questions: “Activity Checklist / Pre-screening completed?” and “10 CFR 50.59 Evaluation completed?”.
- Revised example form Complex Troubleshooting Failure Mode/Cause Table Example (Attachment 5):
  - Updated Problem Statement example for clarity.
  - Moved “Sys ENG: OPEN and MM: OPEN” from row 4 to row 3 to update example form.
  - Revised Troubleshooting Flowchart to update Step 3.7.1 for consistency with revised instructions.

Functional Area Manager: Manager Nuclear Maintenance

INFORMATION USE

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Troubleshooting Sheet

MA-AA-103 - Attachment 2 **Page 1 of 3**

CR Number	Risk ☐ I-High ☐ II-Medium ☐ III-Low ☐ IV-No	Rigor Category ☐ A ☐ B ☐ C ☐ D
Work Authorization (CR/WO)	System	Is a Complex Troubleshooting Plan required? ☐ Yes ☐ No
Component ID		
Operating Conditions		
Initial Problem Statement		

Name of Personnel Having Knowledge of the Problem	Department	Phone Number

Troubleshooting Team Members

- ☐ Engineering ____________________
- ☐ Operations ____________________
- ☐ Maintenance ____________________
- ☐ Vendor ____________________
- ☐ Corporate ____________________
- ☐ O&P ____________________
- ☐ Project Manager ____________________
- ☐ Other ____________________

Operations to determine the following: a. Troubleshooting will cause TS Equipment to become inoperable? ☐ Yes ☐ No b. **IF** Yes, **THEN VERIFY** opposite train equipment and associated EDG are operable.	Operations (Initials)

Form No. 730600(Dec 2009)

relatively recent addition, (see last page), person who relys on past knowledge will select an

Troubleshooting Sheet

MA-AA-103 - Attachment 2 **Page 2 of 3**

Describe the troubleshooting actions or steps for which approval is being requested. Include any initial observations and responses completed by the Operating crew.

Troubleshooting Limits or Boundaries

Describe the equipment configuration during the troubleshooting (extent of equipment isolated, removed from service, made operable, in bypass, controller in manual, etc.) to bound the effects of the troubleshooting and prevent creating an undesirable or unanalyzed equipment configuration. (Refer to MA-AA-103 Attachment 1 for additional risk and rigor consideration.)

Identify the Impact of the Troubleshooting on Plant Equipment (Alarms, Lost Indication, Lost Function, system flow changes, affects on adjacent equipment/systems, potential to affect reactivity by isolation of feedwater heating/control rod movement/boron dilution change or other means, etc. (Refer to MA-AA-103 Attachment 1 for additional risk and rigor consideration.)

Troubleshooting Sheet

MA-AA-103 - Attachment 2 **Page 3 of 3**

Describe the expected results		
Identify any decision or stop points to evaluate progress or subsequent actions		
Activity Checklist/Pre-screening completed? ☐ Yes ☐ No	10 CFR 50.59 Evaluation completed? ☐ Yes ☐ No	
10 CFR 50.59 Screening completed? ☐ Yes ☐ No	PRA Risk evaluated by Operations or O&P? ☐ Yes	
FSRC review required? ☐ Yes ☐ No		
Troubleshooting Team Lead (TTL) Approval, if required (Print Name)	Troubleshooting Team Lead (TTL) Approval, if required (Signature)	Date
Maintenance Manager/Designee Review/Approval (Mark N/A if Rigor Category C or D) (Print Name)	Maintenance Manager/Designee Review/Approval (Mark N/A if Rigor Category C or D) (Signature)	Date
Plant Manager (Nuclear) Approval, if required (Print Name)	Plant Manager (Nuclear) Approval, if required (Signature)	Date
FSRC Chair Approval, if required (Print Name)	FSRC Chair Approval, if required (Signature)	Date
Shift Manager Approval (Print Name)	Shift Manager Approval (Signature)	Date
Results Attained		
Follow-up Action Required		
Additional sheets attached? ☐ Yes ☐ No		
Worker (Print Name)	Worker (Signature)	Date

Page 1 of 4

Dominion®

Procedure or Guidance and Reference Document Approval

1. Document Number: MA-AA-103	2. Revision: 3	3. Document Type: ☒ Admin Procedure ☐ Tech Procedure ☐ GARD

4. Title: Conduct of Troubleshooting

5. Requestor(s) Print Name(s) / Locations: W. R. (Bill) Taylor/NAPS	6. Date: 4-3-09	7. Requestor Phone: NAPS x-2701

8. Document Request: ☐ New ☒ Revision ☐ Cancel ☐ Supersede ☐ Temporary

9. Applicable Nuclear Station(s): Kewaunee ☐ Millstone ☐ North Anna ☒ Surry ☐

10. Reason and Brief Description of Change:

Revised in response to NAPS CA089553 from ACE014023 to add protected train equipment evaluation to troubleshooting sheet:

- Added 5.4.11 "NAPS CA089553, Add protected train equipment evaluation to troubleshooting sheet (from ACE014023)" to update references.
- Revised Troubleshooting Sheet (Attachment 2) in response to NAPS CA089553 to add to page 1 Operations determination of:
  - Troubleshooting will cause TS equipment to become inoperable? Yes or No.
  - IF Yes, THEN VERIFY opposite train equipment and associated EDG are operable.
  - Added Operations (Initials).

11. Change Management: ☐ Change Management Plan Attached ☒ Change Management Plan Not Required

12. Level of Use: ☐ Continuous Use ☐ Reference Use ☒ Information Use ☐ Multiple Use

Fleet Approval

13. Fleet Approval Required by: (Check one box only. Enter Peer Group Name, if applicable)
☒ MAINT Peer Group OR ☒ Functional Area Manager (FAM)

14. Printed Approver Name: S. Heard J.Rigatti for per telecon	15. Signature: [signature]	16. Date: 4/20/09

Site Approval

17. Implementation Prerequisites: (Items in addition to those listed on Document Traveler or Change Management Plan)

None

18. Implementation Prerequisites Reviewed - Procedure Supervisor Signature: John W Plosel	19. Date: 5/14/09

20. Check Nuclear Station(s) for Which Document is being Approved for Implementation.
Kewaunee ☐ Millstone ☐ North Anna ☒ Surry ☐

21. Site Approval (Print Name of FAM): W. Anthes / R. Scanlan for WA	22. Signature: [signature] for WA	23. Date: 5/14/09
24. Facility Safety Review Committee Required? ☒ No ☐ Yes	25. Facility Safety Review Committee (Site) Print Name/Signature: N/A	26. Date: N/A

NOTE: The individual(s) posting a new or revised document to EDMS are responsible for ensuring Nuclear E-Forms is updated.

27. Nuclear E-Forms Updated for Site(s)? ☐ KW ☐ MP ☒ NA ☐ SU ☐ N/A	28. Nuclear E-Forms Updated Print Name/Signature: J. H. Horton / [signature]	29. Date: 5-14-09

30. Document Number: MA-AA-103	31. Revision: 3	32. Effective Date: 5-14-09	33. Expiration Date: None

Key: GARD-Guidance and Reference Document, EDMS-Electronic Document Management System

Form No. 728620 (July 2008)

20. G2.2.39 095/MODIFIED/NAPS: 2918/H/3/3.9/4.5//

Technical Specifications have a note which prohibits starting an RCP with any RCS Cold-leg temperature less than or equal to _____ unless the secondary side water temperature of each SG is less than or equal to 50°F above each of the RCS Cold-leg temperatures.
The bases of this requirement is to ___________.

A. 180°F ; limit rate of change of temperature to restrict stresses caused by thermal gradients.

B. 180°F ; prevent a low temperature overpressure event due to a thermal transient.

C. 280°F ; limit rate of change of temperature to restrict stresses caused by thermal gradients.

D. 280°F ; prevent a low temperature overpressure event due to a thermal transient.

a. Incorrect. First part incorrect but plausible since LTOP has two setpoints and this is one of them, also considering there are two specs 3.4.6 and 3.4.7, the candidate who lacks detailed TS knowledge may errroneously eliminate the higher temperature as a distractor since RCP starts are normally done in Mode 5 and 280°F is mode 4. Plausible since there are restrictions based on temperature change but these restriactions and bases are enveloped by TS 3.4.3, not this TS. It should also be noted that the restriction although ensuring TS 3.4.12 assumptions and analysis are maintained does not appear in the LTOP spec (only discussed in the bases), this is again why a candidate without detailed knowledge might erroneously consider b & d to be distractor and eliminate them as choices.

b. Incorrect. First part incorrect but plausible as noted above. Second part is correct as described in TS 3.4.7 BASES.

c. Incorrect. First part is correct as per TS 3.4.7. Second part incorrect but plausible as discussed in Distractor a.

d. Correct. First part is correct as per TS 3.4.6 & 7. Second part is correct as described in TS 3.4.7 BASES.

Equipment Control

Knowledge of less than or equal to one hour Technical Specification action statements for systems.
(CFR: 41.7 / 41.10 / 43.2 / 45.13)

Tier: 3

Technical Reference: TS 3.4.7 and 3.4.12 and TS Bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops—MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of one steam generator (SG) shall be ≥ 17%.

--------------------------- NOTE ---------------------------

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ 280°F unless the secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures.

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

----

answer
see distractor for analysis of discussion of plausibility of distractors/alternatives.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water level to within limits.	Immediately Immediately
B. Required SG with secondary side water level not within limits. AND One RHR loop OPERABLE.	B.1 Initiate action to restore a second RHR loop to OPERABLE status. OR B.2 Initiate action to restore required SG secondary side water level to within limits.	Immediately Immediately
C. No required RHR loops OPERABLE. OR Required RHR loop not in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is $\geq$ 17% in required SG.	12 hours
SR 3.4.7.3	-------------------NOTE-------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to the required RHR pump not in operation.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops—MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

See 3.4.7-3 for answer

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining a SG with secondary side water level of at least 17% using narrow range instrumentation to provide an alternate method for decay heat removal via natural circulation (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops–MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or a SG with secondary side water level ≥ 17% using narrow range instrumentation and the associated loop isolation valves open. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to provide redundancy for heat removal. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is a SG with its secondary side water level ≥ 17% using narrow range instrumentation. Should the operating RHR loop fail, the SG could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit pump swap operations and tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the pump swap or test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

(continued)

BASES

LCO (continued)

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to meet the SDM of LCO 3.1.1, therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

answer

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

See B 3.4.3-3 for distractor

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 280°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops with circulation provided by an RCP.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

BASES

APPLICABILITY

In MODE 5 with the unisolated portion of the RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be ≥ 17% with the associated loop isolation valves open.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

If all RCS loops are isolated, an SG cannot be used for decay heat removal and RCS water inventory is substantially reduced. In this circumstance, LCO 3.4.8 applies.

ACTIONS

A.1, A.2, B.1, and B.2

If one RHR loop is OPERABLE and the required SG has secondary side water level < 17%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water level. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1 and Note 4, or if no required RHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least one SG is OPERABLE by ensuring its secondary side narrow range water level is ≥ 17% ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that the required RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required RHR pump. If secondary side water level is ≥ 17% in at least one SG, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

BASES

REFERENCES

1. NRC Information Notice 95-35, Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation.

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

supports distractor for plausibility

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

The reactor vessel beltline is the most limiting region of the reactor vessel for the determination of P/T limit curves. The P/T curves include a correction for the difference between the pressure at the point of measurement (hot leg or pressurizer) and the reactor vessel beltline. The P/T limits include instrument uncertainties for pressure and temperature.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

21. G2.3.14 096/NEW//H/3/3.4/3.8//

Unit 2 was initially at 100% power.

A large-break LOCA has occurred on Unit 2.

Which ONE of the following Emergency Ventilation fans **CAN NOT** be used when placing the MCR/ESGR Emergency Ventilation system on Turbine Building Supply, **AND** includes the basis for this restriction?

A. 1-HV-F-41 ; due to the location of the air intake

B. 1-HV-F-41 ; this fan is not seismicly qualified

C. 2-HV-F-41 ; due to the location of the air intake

D. 2-HV-F-41 ; this fan is not seismicly qualified

a. Correct. Per TS 3.7.10 bases both, the fan and the bases are correct. Once again the candidate must have detailed knowledge of the bases (because of the proximaty to Vent Stack B this fan is not used because of the radiological hazards in the event of a release via the vent stack) in order to discriminate a difference from either the choice of fan, or the reason for the choice.

b. Incorrect. First part is correct as noted above. Seecond part is incorrect but plausible since NAPS has had a few issues of seismic qualification as related to operability in the recent past, thus the candidate who lacks detailed knowledge of the basis may default to this distractor.

c. Incorrect. First part is incorrect but plausible since as previously noted the candidate who lacks detailed knowledge of the TS bsaes will not be able to discriminate a difference between the two choices of components. Second part is correct as noted in answer a.

d. Incorrect. First part is incorrect as discussed in Distractor c; second part is also incorrect but plausible as discussed in distractor b.

Radiation Control

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
(CFR: 41.12 / 43.4 / 45.10)

Tier: 3

Technical Reference: TS 3.7.10 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: This is a Unit differences question.

3.7 PLANT SYSTEMS

3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)

LCO 3.7.10 Two MCR/ESGR EVS trains shall be OPERABLE.

----------------------------- NOTE -----------------------------
The MCR/ESGR envelope boundary may be opened intermittently under administrative control.

See Basis Section for answer

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of recently irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MCR/ESGR EVS train inoperable for reasons other than Condition B.	A.1 Restore MCR/ESGR EVS train to OPERABLE status.	7 days
B. One or more required MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR envelope boundary in MODES 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure MCR/ESGR envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>	
	B.3 Restore MCR/ESGR envelope boundary to OPERABLE status.	90 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODES 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time for Condition A not met during movement of recently irradiated fuel assemblies.	D.1.1 Isolate the MCR/ESGR envelope normal ventilation. AND D.1.2 Place OPERABLE EVS train in emergency (outside filtered air supply) mode. OR D.2 Suspend movement of recently irradiated fuel assemblies.	Immediately 1 hour Immediately
E. One or more required MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR envelope boundary during movement of recently irradiated fuel assemblies.	E.1 Suspend movement of recently irradiated fuel assemblies.	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued) OR Two required MCR/ESGR EVS trains inoperable during movement of recently irradiated fuel assemblies for reasons other than Condition B.		
F. Two required MCR/ESGR EVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each required MCR/ESGR EVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.10.2	Perform required MCR/ESGR EVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Not Used	

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4	Perform required MCR/ESGR Envelope unfiltered air inleakage testing in accordance with the MCR/ESGR Envelope Habitability Program.	In accordance with the MCR/ESGR Envelope Habitability Program

B 3.7 PLANT SYSTEMS

B 3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)

BASES

BACKGROUND

The MCR/ESGR Emergency Ventilation System (EVS) provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The MCR/ESGR EVS consists of four 100% capacity redundant trains (2 per unit) that can filter and recirculate air inside the MCR/ESGR envelope or supply filtered makeup air to the MCR/ESGR envelope, and a MCR/ESGR boundary that limits the inleakage of unfiltered air. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan (Ref. 1). Ductwork, valves, dampers, doors, barriers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying outside filtered air. In the event of a Safety Injection (SI), the two MCR/ESGR EVS trains on the accident unit actuate automatically in recirculation. All available trains of MCR/ESGR EVS start automatically on a fuel building radiation monitor signal or manual actuation of the MCR/ESGR Isolation Actuation Instrumentation. These trains can also be aligned to provide filtered outside air when appropriate. Either train from the other unit can be manually actuated to provide filtered outside air approximately 60 minutes after the event. However, due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.10. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42) are required for independence and redundancy.

The MCR/ESGR envelope is the area within the confines of the MCR/ESGR envelope boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The MCR/ESGR envelope is protected during normal operation, natural

(continued)

BASES

BACKGROUND (continued)

events, and accident conditions. The MCR/ESGR envelope boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the MCR/ESGR envelope. The OPERABILITY of the MCR/ESGR envelope boundary must be maintained to ensure that the inleakage of unfiltered air into the MCR/ESGR envelope will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to MCR/ESGR envelope occupants. The MCR/ESGR envelope and its boundary are defined in the MCR/ESGR Envelope Habitability Program.

Upon receipt of an actuating signal(s) (i.e., SI, fuel building radiation monitors or manual), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, and at least two trains of MCR/ESGR EVS receive a signal to actuate to recirculate air in the MCR/ESGR envelope. Approximately 60 minutes after actuation of the MCR/ESGR Isolation Actuation Instrumentation, a single MCR/ESGR EVS train is manually actuated or aligned to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers. The demisters remove any entrained water droplets present, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Although not assumed in the Analysis of Record, pressurization of the MCR/ESGR envelope minimizes infiltration of unfiltered air through the MCR/ESGR envelope boundary from all the surrounding areas adjacent to the MCR/ESGR envelope boundary.

Redundant MCR/ESGR EVS supply and recirculation trains provide the required filtration of outside air should an excessive pressure drop develop across the other filter train.

(continued)

BACKGROUND (continued)

The MCR/ESGR EVS is designed in accordance with Seismic Category I requirements. Any of the actuation signal(s) will isolate the MCR/ESGR envelope and start the MCR/ESGR EVS trains for the affected unit in recirculation. Requiring two of the three MCR/ESGR EVS trains provides redundancy, assuring that at least one train is available to be realigned to provide filtered outside air.

The MCR/ESGR EVS is designed to maintain a habitable environment in the MCR/ESGR envelope for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms.

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EVS provides airborne radiological protection for the MCR/ESGR envelope occupants, as demonstrated by the MCR/ESGR envelope accident dose analyses for the most limiting DBA (LOCA) fission product release presented in the UFSAR, Chapter 15 (Ref. 2). The accident analysis assumes that at least one train is aligned to provide filtered outside air to the MCR/ESGR envelope approximately 60 minutes after MCR/ESGR envelope isolation, but does not take any credit for automatic start of the trains in the recirculation mode or any filtration of recirculated air. Since the MCR/ESGR EVS train associated with 1-HV-F-41 can not be used to provide filtered outside air (due to the location of its air intake with respect to Vent Stack B), it can not be used to satisfy the requirements of LCO 3.7.10.

The North Anna UFSAR describes potentially hazardous chemicals stored onsite in quantities greater than 100 lb. These include hydrogen, sulfuric acid, sodium hydroxide, hydrazine, ethanolamine, and sodium hypochlorite. Evaluations for accidental release of these chemicals indicate that the worst-case concentrations at the control room intake would be expected to be less than their

(continued)

APPLICABLE SAFETY ANALYSES (continued)

respective toxicity limit (Refs. 1 and 4). The assessment assumed no action being taken by the control room operator (i.e., normal or emergency supply system remains operating).

In the event of fire/smoke external to the MCR/ESGR envelope, equipment and procedures are available to maintain habitability of the control room. Smoke detectors are installed in the return ducts to the MCR Air-Handling Units (AHUs), in the near vicinity of the ESGR AHUs, and in the MCR/ESGR EVS supply ducts, as well as other numerous locations in the ESGRs and MCR. Smoke detectors are also installed in the MCR/ESGR chiller rooms, which are ventilated with air from the Turbine Building, and the Mechanical Equipment rooms. If smoke is detected, the MCR/ESGR normal and EVS supply can be manually isolated. The fire response procedures provide direction for removing smoke from the MCR or ESGRs. (Ref. 5)

For the remainder of the DBAs, MCR/ESGR envelope isolation is not assumed. Normal ventilation with 500 cfm of additional inleakage is assumed. The safety analysis for a fuel handling accident (FHA) assumes isolation of the MCR/ESGR envelope.

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one train is available to be manually aligned to provide outside filtered air to the MCR/ESGR envelope, if a single active failure disables one of the two required OPERABLE trains. Total system failure, such as from a loss of both required EVS trains or from an inoperable MCR/ESGR envelope boundary, could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms, in the event of a large radioactive release.

(continued)

BASES

LCO (continued)

The MCR/ESGR EVS is considered OPERABLE when the individual components necessary to limit MCR/ESGR envelope occupant exposure are OPERABLE in the two required trains of the MCR/ESGR EVS. 1-HV-F-41 can not be used to satisfy the requirements of LCO 3.7.10.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

In order for the MCR/ESGR EVS trains to be considered OPERABLE, the MCR/ESGR envelope boundary must be maintained such that the MCR/ESGR envelope occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that MCR/ESGR envelope occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the MCR/ESGR envelope boundary to be opened intermittently under administrative controls. This Note only applies to openings in the MCR/ESGR envelope boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the MCR/ESGR envelope. This individual will have a method to rapidly close the opening and restore the MCR/ESGR envelope boundary to a condition equivalent to the design condition when a need for MCR/ESGR isolation is indicated.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, MCR/ESGR EVS must be OPERABLE to ensure that the MCR/ESGR envelope will remain habitable during and following a DBA.

The MCR/ESGR EVS must be OPERABLE to respond to the release from a FHA involving recently irradiated fuel assemblies. The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving recently irradiated fuel assemblies (i.e., fuel assemblies that have occupied part of a critical reactor core within the previous 300 hours) due to radioactive decay.

ACTIONS

A.1

When one required MCR/ESGR EVS train is inoperable, for reasons other than an inoperable MCR/ESGR envelope boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS train is adequate to perform the MCR/ESGR envelope occupant protection function. However, the overall reliability is reduced because a failure in the required OPERABLE EVS trains could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the MCR/ESGR envelope boundary and into the MCR/ESGR envelope can result in MCR/ESGR envelope occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem total effective dose equivalent), or inadequate protection of MCR/ESGR envelope occupants from hazardous chemicals or smoke, the MCR/ESGR envelope boundary is inoperable. Actions must be taken to restore an OPERABLE MCR/ESGR envelope boundary within 90 days. During the period that the MCR/ESGR envelope boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on MCR/ESGR envelope occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that MCR/ESGR envelope occupant

(continued)

BASES

ACTIONS

B.1 (continued)

radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that MCR/ESGR envelope occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable MCR/ESGR envelope boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of MCR/ESGR envelope occupants within analyzed limits while limiting the probability that MCR/ESGR envelope occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the MCR/ESGR envelope boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable required MCR/ESGR EVS train or the inoperable MCR/ESGR envelope boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1.1, D.1.2, and D.2

During movement of recently irradiated fuel, if the inoperable MCR/ESGR EVS train cannot be restored to OPERABLE status within the required Completion Time, the MCR/ESGR envelope must be isolated immediately and the remaining OPERABLE MCR/ESGR train placed in service within one hour. These actions will ensure that the MCR/ESGR envelope is in a configuration that would protect the occupants from radioactive exposure consistent with the DBA assumptions and ensure that any active failures would be readily detected.

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

During movement of recently irradiated fuel assemblies, if a required train of MCR/ESGR EVS train becomes inoperable due to an inoperable MCR/ESGR envelope boundary or two required MCR/ESGR EVS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

F.1

When two required MCR/ESGR EVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR envelope boundary (i.e., Condition B), the MCR/ESGR EVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the one train redundancy.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

Not Used

SR 3.7.10.4

This SR verifies the OPERABILITY of the MCR/ESGR envelope boundary by testing for unfiltered air inleakage past the MCR/ESGR envelope boundary and into the MCR/ESGR envelope. The details of the testing are specified in the MCR/ESGR Envelope Habitability Program. The MCR/ESGR envelope is considered habitable when the radiological dose to MCR/ESGR envelope occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the MCR/ESGR envelope occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the MCR/ESGR envelope is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the MCR/ESGR envelope boundary to OPERABLE status provided mitigating actions can ensure that the MCR/ESGR envelope remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 6) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 7). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 8). Options for restoring the MCR/ESGR envelope boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the MCR/ESGR envelope boundary, or a combination of these actions.

(continued)

BASES

SR 3.7.10.4 (continued)

Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the MCR/ESGR envelope boundary has been restored to OPERABLE status.

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Chapter 15.
3. 10 CFR 50, Appendix A.
4. Control Room Habitability Study (Supplement to 1980 Onsite Control Room Habitability Study - North Anna Power Station Units 1 and 2, January 1982.
5. Letter from L.N. Hartz (Virginia Electric and Power Company) to the USNRC, dated March 3, 2004, Response to Generic Letter 2003-01, "Control Room Habitability - Control Room Testing & Technical Information."
6. Regulatory Guide 1.196.
7. NEI 99-03, "Control Room Habitability Assessment," June 2001.
8. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694)

22. G2.3.6 097/NEW//H/4/2.0/3.8//SIMILAR 2008 Q.98

The Waste Gas Decay Tanks (WGDT) were sampled to determine the quantity of radioactive material contained within each tank.

Confirmed sample results are as follows:

- "A" WGDT - 15,000 curies of noble gas.
- "B" WGDT - 6,000 curies of noble gas.

Based on these sample results, which ONE of the following identifies the implications per TR 3.1.3, Gas Storage Tanks, **AND** includes the associated TRM BASES?

The quantity of radioactive material in the WGDTs ____________. The TRM bases for restricting the quantity of radioactive material is to limit dose at the exclusion boundary to ____________.

A. is within limits, entry into TR 3.10.3 action is NOT required ; 0.5 rem.

B. is within limits, entry into TR 3.10.3 action is NOT required ; 0.1 rem.

C. is NOT within limits, enter TR 3.10.3 action ; 0.5 rem.

D. is NOT within limits, enter TR 3.10.3 action ; 0.1 rem.

a. Correct. First part is correct, 25,000 is the limit so we are within it. The second part is also correct per the TR bases.

b. Incorrect. First part is correct as noted above. Second part is incorrect but plausible since the candidate who lacks detailed knowledge of the bases may erroneously assume that a limit of .5 would be associated with areas of the plant and thus the exclusion area boundary would have to be lower.

c. Incorrect. Plausible since this value is well in excess of those that would normally ever be encountered and the candidate may be aware that there is a limit, but if they are not sure of the specific value they may default to this distractor based on the large values given; second part is correct.

d. Incorrect. First part is incorrect as discussed in Distractor c; second part is also incorrect but plausible as discussed in distractor b.

QUESTIONS REPORT
for NAPS 2010 SRO NRC Exam rev3

Radiation Control

Ability to approve release permits.
(CFR: 41.13 / 43.4 / 45.10)

Tier: 3

Technical Reference: TR 3.10.3 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: Operations is not in the line of authority for approval of release permits at NAPS. This test item was written to meet the intent of the K/A by providing an equivalent evaluation of candidate knowledge as it relates to radioactive storage and the like. The candidate must know both the limits and their bases.

3.10 RADIOACTIVE STORAGE

3.10.3 Gas Storage Tanks

TR 3.10.3 The quantity of radioactive material contained in each gas storage tank shall be ≤ 25,000 curies of noble gases (considered as Xe-133).

answer to first part - see Bases for 2nd part.

APPLICABILITY: At all times.

ACTIONS

NOTES

1. The provisions of TR 3.0.3 are not applicable.
2. Separate Condition entry is allowed for each tank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Quantity of radioactive material in gas storage tank not within limit.	A.1 Suspend all additions of radioactive material to the affected tank.	Immediately
	<u>AND</u>	
	A.2 Reduce the quantity of radioactive material in the affected tank to within limit.	48 hours

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.10.3.1 Verify quantity of radioactive material contained in each gas storage tank is ≤ 25,000 curies of noble gases (considered as Xe-133).	31 days AND ------NOTE------ Not required to be performed if Reactor Coolant System specific activity DOSE EQUIVALENT I-131 is ≤ 1.0 μCi/gm ---------------- Once per 24 hours when radioactive materials are being added to the tank

B 3.10 RADIOACTIVE STORAGE

B 3.10.3 Gas Storage Tanks

BASES

The tanks included in this TR are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another TR or Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion boundary will not exceed 0.5 rem in an event of 2 hours.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled released of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

23. G2.4.26 098/NEW//H/3/3.1/3.6//

Both units are at 100% power with Operations shift staffing at the minimum required by OP-AA-100, Conduct of Operations.

During the response to a fire, the scene leader is overcome by smoke, and is transported offsite for medical attention. The only other qualified scene leader on site is a licensed RO, who is currently assigned Backboards.

Which ONE of the following identifies the impact on shift staffing, **AND** includes the action required?

A. Fire brigade minimum manning requirements are NOT met; immediately initiate action to restore the required fire brigade manning within 1 hour.

B. Fire brigade minimum manning requirements are NOT met; immediately initiate action to restore the required fire brigade manning within 2 hours.

C. Fire brigade minimum manning requirements are NOT met; within 1 hour, initiate action to restore the required fire brigade manning within 2 hours.

D. Assign the licensed RO to assume scene leader duties, and reassign Backboards duties to a non-licensed operator; Fire brigade and Shift minimum manning requirements are met.

a. Incorrect. First part is correct (see b). Second part is incorrect but plausible if applicant believes manning must be restored within one hour (typical with many TS & TR actions).

b. Correct. Fire brigade scene leader is supplied by OPS, and the fire brigade members are NOT part of the minimum shift crew (although they may be assigned other duties besides fire brigade). Per TR 7.3, Condition A, immediate action is required to restore fire brigade manning within 2 hrs.

c. Incorrect. First part is correct (see b). Second part is incorrect but plausible as discussed in Distractor a (candidate could easily assume they have up to an hour to take action).

d. Incorrect. First part is an option, but if the re-assignment were made then the shift manning requirements would not be met as stated in the rest of the distractor (see b). Plausible if candidate confuses the "normal" vs "minimum" manning requirements of OP-AA-100.

Emergency Procedures / Plan

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.
(CFR: 41.10 / 43.5 / 45.12)

Tier: 3

Technical Reference: OP-AA-100, Conduct of Operations, (Att. 2, pg. 15 of 31); TR 7.3 Fire Brigade; TS 5.2.2, Unit Staff

Proposed references to be provided to applicants during examination: None

Learning Objective: vision #13559

Question History:

additional info:

OBJ 13559; VISION item #5788; 624.26

7.0 FIRE PROTECTION

7.3 Fire Brigade

TR 7.3 A Fire Brigade of at least 5 members shall be maintained onsite.

- - - - - - - - - - NOTES - - - - - - - - - -

1. The Fire Brigade Scene Leader and at least two brigade members shall have sufficient training in or knowledge of safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability.

2. The Fire Brigade shall not include the minimum shift crew required by Technical Specification Section 5.2.2 or any personnel required for other essential functions during a fire emergency.

- -

APPLICABILITY: At all times.

ACTIONS

- - - - - - - - - - NOTE - - - - - - - - - -

TR 3.0.3 is not applicable.

- -

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fire Brigade minimum manning requirements not met due to unexpected absence.	A.1 Initiate action to restore required Fire Brigade manning. <u>AND</u> A.2 Restore required Fire Brigade manning.	Immediately 2 hours
B. Required Actions and associated Completion Times not met.	B.1 Submit a Plant Issue.	Immediately

B 7.0 FIRE PROTECTION

B 7.3 Fire Brigade

BASES

None.

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. An auxiliary operator shall be assigned to each reactor containing fuel and an additional auxiliary operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

   Two unit sites with both units shutdown or defueled require a total of three auxiliary operators for the two units.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Deleted

5.2 Organization

5.2.2 Unit Staff (continued)

e. The operations manager shall hold (or have previously held) a Senior Reactor Operator License for North Anna or a similar design Pressurized Water Reactor plant. The Supervisor Nuclear Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station.

f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Attachment 2
(Page 15 of 31)
Shift Operations

North Anna

North Anna Normal Requirements	
With Both Units in Mode 1, 2, 3, 4, 5, or 6	With either unit defueled
SM-1	SM-1
US-2	US-1
RO-4[c]	RO-3
AO-4	AO-4
STA-1	STA-1
Communicator-2[d]	Communicator-2[d]
Fire Brigade-5[a]	Fire Brigade-5[a]
First Aid-2[b]	First Aid-2[b]

a. A minimum of 3 from Operations, one of whom is a qualified Scene Leader (Security typically provides the remaining members). The Operators are not included in the RO/AO manning and may have other duties.
b. Coordinated by Security.
c. Fourth RO may have other duties.
d. May have other duties in addition to being a communicator.

In the event that actual shift staffing falls below the normal requirements, refer to Technical Specifications and the Technical Requirements Manual.

North Anna Minimum Requirements	
With Both Units in Mode 1, 2, 3, 4, 5, or 6	With either unit defueled
SM-1	SM-1
US-1	US-0
RO-3[c]	RO-3
AO-4	AO-3
STA-1	STA-1
Communicator-2[c]	Communicator-2[c]
Fire Brigade-5[a]	Fire Brigade-5[a]
First Aid-2[b]	First Aid-2[b]

a. A minimum of 3 from Operations, one of whom is a qualified Scene Leader (Security typically provides the remaining members). The Operators are not included in the RO/AO manning and may have other duties.
b. Coordinated by Security.
c. May have other duties.

Provide a copy of the TRM and Tech Specs

Unit-1 is operating in Mode 5 with RHR in service. Unit-2 is operating at 100% power. Shift manning is as follows:

- One (1) Shift Manager
- Two (2) Unit Supervisors
- Four (4) Reactor Operators
- Six (6) Non-licensed Operators
- One (1) Shift Technical Advisor

During the shift, the running Unit-1 RHR pump catches fire. The fire brigade is dispatched and the running RHR pump is secured. The standby RHR pump trips immediately after starting and the Unit-1 crew initiates 1-AP-11, "Loss of RHR."

As fire fighting efforts proceed, an operator and a security officer assigned to the fire brigade are overcome by smoke inhalation.

Given the present situation, the minimum __________ manning requirements established by __________ are not met and must be restored __________.

A. fire brigade; TRM; within 2 hours

B. shift; OP-AA-100; within 2 hours

C. shift; TRM; immediately

D. fire brigade; OP-AA-100; immediately

Answer: A

24. WE04EA2.1 099/BANK/NAPS: 7359/H/3/3.4/4.3/3/

Given the following conditions:

- Unit 1 was initially at 100% power.
- PRZR pressure and level began decreasing rapidly, and the crew manually tripped the reactor.
- Safety Injection automatically actuated.

The following conditions exist:

- Containment pressure - 11.1 psia and stable.
- Containment gaseous, particulate, and area radiation monitors - stable at pre-event values.
- 1-VG-RI-180-1, MGP Vent Stack B Rad Monitor - HI alarm LIT.
- SG pressures - 1000 psig and stable.

The crew is performing 1-E-0, Reactor Trip or Safety Injection.

Which ONE of the following identifies the procedure the operators will transition to when leaving 1-E-0?

A. 1-E-1, Loss of Reactor or Secondary Coolant

B. 1-ES-1.2, Post-LOCA Cooldown and Depressurization

C. 1-ECA-1.1, Loss of Emergency Coolant Recirculation

D. 1-ECA-1.2, LOCA Outside Containment

a. Incorrect. Plausible since plant conditions clearly support that a LOCA is in progress. E-1 has diagnostic steps to transition the operator to ECA-1.2, and even if the candidate diagnoses that this is an ECA-1.2 scenario they may lack detailed knowledge and assume that E-1 would be entered first and then send you to ECA-1.2.

b. Incorrect. Plausible since E-0 does have transitional step to this procedure. For the plant conditions cooldown and depressurization is a potential stratgey, but this transition would not yield the desired result, however the candidate who lacks detailed knowledge or incorrectly analyzes the event may select this distractor.

c. Incorrect. Plausible since this is a possible transition & the conditions support a loss of RCS inventory outside containment which ECA-1.1 was specifically written to mitigate. There is a transition to ECA-1.1 for the event, but ECA-1.2 would be implemented first.

d. Correct. Based on the plant conditions the operator will transition to ECA-1.2 from E-0 Step 14 (again the candidate must have detailed knowledge of the procedure and properly diagnose the given conditons in order to possitively conclude that this choice is correct).

LOCA Outside Containment

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) (CFR: 43.5 / 45.13)

Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Tier: 1
Group: 1

Technical Reference: E-0, E-1, ECA-1.2 & WOG Background Document

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info: Failure to implement the correct flow path would result in delaying required actions and cause larger exposure and spread contamination.

NORTH ANNA POWER STATION

EMERGENCY PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-E-0	REACTOR TRIP OR SAFETY INJECTION (WITH TEN ATTACHMENTS)	42 PAGE 1 of 21

PURPOSE

This procedure provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a Reactor trip or Safety Injection, to assess plant conditions, and to identify the appropriate recovery procedure.

TIME CRITICAL ACTIONS

ENTRY CONDITIONS

1) The following are symptoms that require a Reactor trip, if one has not occurred:
   - A Reactor protection system setpoint has been exceeded
   - A Turbine protection system setpoint with power greater than P-8 setpoint
2) The following are symptoms of a Reactor trip:
   - Any Reactor trip first out Annunciator - LIT
   - Reactor Trip and Bypass Breakers - OPEN
   - Rod Bottom Lights - LIT
   - Neutron flux - DECREASING
3) The following are symptoms that require a Reactor trip and Safety Injection, if one has not occurred:
   - Low PRZR pressure
   - High Containment pressure
   - Steamline differential pressure
   - High steamflow with lo-lo Tave
   - High steamflow with low steam pressure
4) The following are symptoms of a Reactor trip and Safety Injection:
   - Any SI first out Annunciator - LIT
   - Any Low-Head SI Pumps - RUNNING
5) Transition from another plant procedure.

UNIT ONE

CONTINUOUS USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]___	VERIFY REACTOR TRIP:	☐ IF Reactor will NOT trip, THEN GO TO 1-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, STEP 1.
	☐ a) Manually Trip Reactor	
	b) Check the following:	
	☐ • Reactor Trip and Bypass Breakers - OPEN	
	☐ • Rod Bottom Lights - LIT	
	☐ • Neutron flux - DECREASING	
[2]___	VERIFY TURBINE TRIP:	
	☐ a) Manually Trip Turbine	
	☐ b) Verify all Turbine Stop Valves - CLOSED	☐ b) Put both EHC Pumps in PTL. ☐ IF Turbine is still NOT tripped, THEN manually run back Turbine. ☐ IF Turbine cannot be run back, THEN close MSTVs and Bypass Valves.
	☐ c) Reset Reheaters	
	☐ d) Verify Generator Output Breaker - OPEN	☐ d) IF Generator Output Breaker does NOT open after 30 seconds, THEN manually open G-12 AND Exciter Field Breaker.

NOTE: Diagnostic steps begin on pg 12

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3]___	VERIFY BOTH AC EMERGENCY BUSSES - ENERGIZED	Do the following:
		☐ a) IF no AC Emergency Bus is energized, THEN immediately restore power to at least one AC Emergency Bus.
		☐ IF power cannot be restored, THEN GO TO 1-ECA-0.0, LOSS OF ALL AC POWER, STEP 1.
		☐ b) Try to restore power to de-energized AC Emergency Bus using 0-AP-10, LOSS OF ELECTRICAL POWER, as time permits.
		☐ Continue with Step 4.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[4]___	CHECK SI STATUS:	
	a) Check if SI is actuated: ☐ • Low-Head SI Pumps - RUNNING <u>OR</u> ☐ • Any SI First-Out Annunciator - LIT	a) Check if SI is required or imminent as indicated by any of the following: ☐ • Low PRZR pressure ☐ • High Containment pressure ☐ • Steamline differential pressure ☐ • High steamflow with either: • Lo-Lo Tave <u>OR</u> • Low steam pressure ☐ <u>IF</u> SI required, <u>THEN</u> GO TO Step 4b. ☐ <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> GO TO 1-ES-0.1, REACTOR TRIP RESPONSE, STEP 1.
	☐ b) Manually actuate SI	
5.___	INITIATE ATTACHMENT 4, EQUIPMENT VERIFICATION, WHILE CONTINUING WITH THIS PROCEDURE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.___	VERIFY SI FLOW:	
	☐ a) VERIFY HIGH-HEAD COLD LEG SI FLOW - INDICATED	a) Verify High-Head flow indicated on the following: ☐ • 1-SI-FI-1943 ☐ • 1-SI-FI-1943-1 OR ☐ • 1-SI-FI-1961 (NQ) ☐ • 1-SI-FI-1962 (NQ) ☐ • 1-SI-FI-1963 (NQ) ☐ IF High-Head flow is NOT indicated, THEN immediately initiate ATTACHMENT 6, MANUAL VERIFICATION OF SI FLOWPATH, to restore High-Head SI flow, while continuing with this procedure.
	☐ b) Check RCS pressure - LESS THAN 225 PSIG [450 PSIG]	☐ b) GO TO Step 7.
	☐ c) Low-Head SI Pump flow - INDICATED	c) Verify Low-Head flow indicated on the following: ☐ • 1-SI-FI-1945 ☐ • 1-SI-FI-1946 ☐ IF NOT, THEN manually start pumps and align valves as necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.___	VERIFY AFW FLOW:	
	☐ a) AFW flow to all SGs- INDICATED	☐ a) Manually align AFW valves and start pumps, as necessary.
	☐ b) Verify total AFW flow - GREATER THAN 340 GPM	☐ b) IF narrow range level is greater than 11% [22%] in any SG, THEN control feed flow to maintain narrow range level AND GO TO Step 8.
		☐ IF narrow range level is less than 11% [22%] in all SGs, THEN manually start pumps and align valves to establish at least 340 gpm AFW flow. IF AFW flow greater than 340 gpm cannot be established, THEN GO TO 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, STEP 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*8. ___	CHECK RCS AVERAGE TEMPERATURE:	
	a) Check Temperature Control: • STEAM DUMPS - CONTROLLING: ☐ • STABLE AT 547°F OR ☐ • TRENDING TO 547°F **OR** • SG PORVs - CONTROLLING: ☐ • STABLE AT 551°F OR ☐ • TRENDING TO 551°F	a) Do the following: IF temperature is less than control value AND decreasing, THEN do the following: ☐ 1) Stop dumping steam. ☐ 2) IF cooldown continues, THEN adjust total AFW flow to maintain greater than 340 gpm until at least one SG narrow range level is greater than 11% [22%]. ☐ 3) IF cooldown continues, THEN close MSTVs and Bypass Valves. ☐ 4) GO TO Step 9.

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8.	CHECK RCS AVERAGE TEMPERATURE: (Continued)	
		IF temperature is greater than control value and increasing, THEN do the following:
		☐ • Dump steam to the Condenser
		OR
		☐ • Dump steam using SG PORVs
		OR
		• Dump steam using Decay Heat Release Valve:
		1) Locally open isolation valve(s) for NON-RUPTURED SG(s) to Decay Heat Release Valve:
		☐ • 1-MS-19, A Steam Line to 1-MS-HCV-104 Non-Return Valve
		☐ • 1-MS-58, B Steam Line to 1-MS-HCV-104 Non-Return Valve
		☐ • 1-MS-96, C Steam Line to 1-MS-HCV-104 Non-Return Valve
		☐ 2) Locally open 1-MS-20, Decay Heat Release Valve Upstream Isolation Valve.
		☐ 3) Manually open 1-MS-HCV-104, Decay Heat Release Valve.
		☐ GO TO Step 8.b.
	☐ b) Adjust total AFW flow between SGs to maintain greater than 340 gpm until at least one SG narrow range level is greater than 11% [22%].	

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 9. ___	CHECK PRZR PORVs AND SPRAY VALVES:	
	☐ a) Check PRZR PORVs - CLOSED	a) IF PRZR pressure less than 2335 psig, THEN manually close PORVs: ☐ • 1-RC-PCV-1455C ☐ • 1-RC-PCV-1456 ☐ IF any PORV cannot be closed, THEN manually close the associated Block Valve. ☐ IF the Block Valve cannot be closed, THEN GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, STEP 1.

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* 9. CHECK PRZR PORVs AND SPRAY VALVES: (Continued)

b) Check PRZR Spray Valves:

☐ • Responding to control PRZR pressure at 2235 psig

OR

☐ • Demand at Zero with Spray Valves - CLOSED

b) IF PRZR pressure less than 2235 psig, THEN manually close valves using controller:

☐ • 1-RC-PCV-1455A

☐ • 1-RC-PCV-1455B

☐ Verify PRZR spray valves closed. IF NOT, THEN place failed valve remote close switch in CLOSE:

☐ • 1-RC-SOV-1455A, 1-RC-PCV-1455A REMOTE CLOSE SOV

☐ • 1-RC-SOV-1455B, 1-RC-PCV-1455B REMOTE CLOSE SOV

IF spray valves can NOT be closed, THEN do the following:

IF 1-RC-PCV-1455A failed open, THEN:

☐ 1) Stop 1-RC-P-1C.

☐ 2) Stop 1-RC-P-1A.

IF 1-RC-PCV-1455B failed open, THEN:

☐ 1) Stop 1-RC-P-1C.

☐ 2) IF 1-RC-P-1A is running, THEN stop 1-RC-P-1B.

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 9.	CHECK PRZR PORVs AND SPRAY VALVES: (Continued)	
☐	c) Check PRZR PORV Block Valves - AT LEAST ONE OPEN	c) Open at least one Block Valve unless both are closed to isolate open or faulty PRZR PORVs:
		☐ • 1-RC-MOV-1536 (1-RC-PCV-1455C)
		☐ • 1-RC-MOV-1535 (1-RC-PCV-1456)
10.___	CHECK RCP TRIP AND CHARGING PUMP RECIRC CRITERIA:	
☐	a) RCS subcooling based on Core Exit TCs - LESS THAN 25°F [85°F]	☐ a) GO TO Step 11.
☐	b) Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS	☐ b) GO TO Step 11.
☐	c) Stop all RCPs	
	d) Check if Charging Pump Recirc Valves should be closed:	
☐	1) RCS pressure - LESS THAN 1275 PSIG [1475 PSIG]	☐ 1) GO TO Step 11.
	2) Close Charging Pump Recirc Valves:	☐ 2) Close 1-CH-MOV-1373, Charging Pump Recirc Header Isolation Valve.
☐	• 1-CH-MOV-1275A	
☐	• 1-CH-MOV-1275B	
☐	• 1-CH-MOV-1275C	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. ___	**CHECK SGs- NOT FAULTED:**	☐ GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, STEP 1.
	☐ • All SG pressures: • GREATER THAN 80 PSIG AND • UNDER CONTROL OF OPERATOR	
12. ___	**CHECK THAT SG TUBES ARE NOT RUPTURED:**	
	☐ a) Level in any SG - INCREASING IN AN UNCONTROLLED MANNER	☐ a) GO TO Step 12c.
	☐ b) GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, STEP 1	
	c) Check Radiation Monitors - NORMAL: ☐ • SG Blowdown radiation last known valid indication ☐ • Condenser Air Ejector radiation last known valid indication ☐ • SG Main Steamline radiation ☐ • Terry Turbine AFW Pump exhaust radiation	☐ c) GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, STEP 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**CHECK IF RCS IS INTACT INSIDE CONTAINMENT:**	☐ GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, STEP 1.
	☐ • Containment pressure - NORMAL	Distractor
	☐ • Containment Recirc Spray Sump level - NORMAL	
	☐ • Containment radiation - NORMAL	
14.	**CHECK FOR OUTSIDE CONTAINMENT INVENTORY LOSS:**	☐ Determine cause of abnormal conditions.
	☐ a) Vent Stack radiation - NORMAL:	☐ IF cause is a loss of RCS inventory outside Containment, THEN GO TO 1-ECA-1.2, LOCA OUTSIDE CONTAINMENT, STEP 1.
	• MGP Vent Stack A	Answer
	AND	
	• MGP Vent Stack B	
	b) Safeguard Area Sump level annunciators - NOT LIT:	see next pg and distractor analysis for discussion of alternative choices
	☐ • Annunciator Panel "A" C-1	
	☐ • Annunciator Panel "E" F-8	
	☐ c) Auxiliary Building Sump level - Annunciator Panel "E" F-6 - NOT LIT	
	☐ d) Ambient Area Temperatures - Annunciator Panel "A" C-4 - NOT LIT	

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. ___	**CHECK IF SI SHOULD BE REDUCED:**	
	☐ a) RCS subcooling based on Core Exit TCs - GREATER THAN 25°F	☐ a) GO TO Step 23.
	b) Secondary heat sink:	☐ b) GO TO Step 23.
	☐ • Total AFW flow to SGs - GREATER THAN 340 GPM	
	OR	
	☐ • At least one SG narrow range level - GREATER THAN 11%	
	☐ c) RCS pressure - STABLE OR INCREASING	☐ c) GO TO Step 23.
	☐ d) PRZR level - GREATER THAN 21%	☐ d) Try to stabilize RCS pressure with normal PRZR spray.
		☐ RETURN TO Step 15a.
16. ___	RESET BOTH TRAINS OF SI	☐ Perform 1-AP-0, RESETTING SI LOCALLY, while continuing with this procedure.
17. ___	STOP ALL BUT ONE CHARGING PUMP AND PUT IN AFTER-STOP	
18. ___	**CHECK RCS PRESSURE - STABLE OR INCREASING**	☐ GO TO 1-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, STEP 1.

Distractor

25. WE05G2.4.20 100/BANK//H/3/3.8/4.3/4/

Given the following conditions:

- Unit 1 tripped from 100% power due to a loss of offsite power.
- Multiple failures have resulted in a loss of all AFW flow.
- Operators have implemented 1-FR-H.1, Loss of Secondary Heat Sink.

The CAUTION prior to Step 2 of 1-FR-H.1 reminds the crew that RCS bleed and feed should be initiated immediately if wide-range level in ____________ decreases to less that 14% because of a loss of heat sink; the basis for this requirement is to _____________.

A. ANY SG ; minimize core uncovery and prevent inadequate core cooling.

B. ANY SG ; minimize challenges to SG tube integrity from creep rupture failure.

C. ANY 2 SGs ; minimize core uncovery and prevent inadequate core cooling.

D. ANY 2 SGs ; minimize challenges to SG tube integrity from creep rupture failure.

a. Incorrect. Plausible since on a trip there is enough SG inventory to last for at least 20 minutes, so the candidate may feel that attempts have been made the outlook is bleak and it wouldn't make sence to wait any longer. Second part is correct IAW the Background document.

b. Incorrect. First part incorrect but plausible as noted above. Second part is incorrect but plausible, there are concerns with creep rupture failure related to inadequate SG level, but these would be at significantly higher temperatures than those indicated by the information provided in the stem.

c. Correct. First part is correct as stated in FR-H.1; second part also correct as stated in the background document.

d. Incorrect. First part is correct as noted in answer c. Second part is incorrect but plausible as explained in distractor b.

Loss of Secondary Heat Sink

Knowledge of the operational implications of EOP warnings, cautions, and notes.
(CFR: 41.10 / 43.5 / 45.13)

Tier: 1
Group: 1

Technical Reference: EOP FR-H.1 and WOG Background document

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question History:

additional info:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If wide range SG level in any two SGs is less than 14% [32%] because of a loss of secondary heat sink, then Step 14 through Step 23 should be performed immediately for RCS bleed and feed.

* *

answer to first part
see Background document
for 2nd part.

2.___ TRY TO ESTABLISH AFW FLOW TO AT LEAST ONE SG:

a) Check SG Blowdown and Sample Isolation:

☐ a) Manually close valves.

☐ • SG Blowdown Isolation Valves - CLOSED

☐ • SG Sample Isolation Valves - CLOSED

See distractor analysis
for discussion of
alternatives.

b) Review Control Room indications to determine cause of AFW failure:

☐ • ECST level

☐ • AFW Pump power supply

☐ • AFW valve alignment

☐ c) Start at least one AFW Pump from the Control Room

c) IF Motor-Driven AFW Pumps cannot be started because of loss of control power, THEN try to start at least one pump as follows, while continuing with this procedure:

☐ • From the Auxiliary Shutdown Panel

☐ • Using 0-MOP-26.11, 4160-VOLT BREAKER LOCAL MANUAL OPERATION

(STEP 2 CONTINUED ON NEXT PAGE)

Naps deviation, this caution located prior to step 2

STEP DESCRIPTION TABLE FOR FR-H.1 Step 3 - CAUTION 1

CAUTION: If parameter (X.01) [(X.02) for adverse containment] is exceeded due to loss of secondary heat sink, RCPs should be tripped and Steps 10 through 16 should be immediately initiated for bleed and feed.

PURPOSE: To alert the operator of the parameter indications which indicate that bleed and feed should be initiated

BASIS:

answer to 2nd part

If the operator cannot restore feedwater flow to the SGs, conditions will degrade to the point where RCS bleed and feed must be established to minimize core uncovery and prevent inadequate core cooling. The parameter and setpoints should be monitored during subsequent steps to reestablish feed flow. An in-depth discussion of this is provided in subsection 2.2, RCS Bleed and Feed Heat Removal, of this background document.

ACTIONS:

Determine if parameter (X.01) [(X.02) for adverse containment] is exceeded due to loss of secondary heat sink

INSTRUMENTATION:

Plant specific instrumentation to indicate status of the loss of secondary heat sink parameters

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

- The importance of establishing bleed and feed as an alternative heat sink to prevent core uncovery and inadequate core cooling.
- If PORV block valves are closed, they should be opened at this time unless they are closed to isolate a faulty PORV.
- When the RCPs are stopped due to loss of heat sink, RCS pressure and temperature are expected to increase slightly and stabilize below the PRZR PORV setpoint. RCS pressure and temperature will continue to be relatively constant until SG dryout occurs (approximately 20 - 30 minutes). At this point, the primary-to-secondary heat transfer rate degrades and the RCS begins to heat up and repressurize and will eventually result in the opening of the PRZR PORVs.

  This should not be confused with the onset of natural circulation in which the RCS pressure continues to increase after the RCPs are stopped and may reach the PRZR PORV setpoint. The key to determining if the RCS pressure rise is due to loss of heat sink or natural circulation is the loop delta-T. The loop delta-T is expected to be large for natural circulation and small for a loss of heat sink since there is no heat transfer to the secondary.

  Therefore, verifying a slowly increasing RCS pressure and temperature trend plus a large loop delta-T prior to the PORV opening confirms natural circulation whereas a relatively stable temperature and pressure and a small loop delta-T combined with SG wide range low level prior to the PORV opening confirms a loss of heat sink.

PLANT-SPECIFIC INFORMATION:

- (X.01) Parameter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy. Refer to Background Document for guideline FR-H.1.
- (X.02) Parameter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy and post accident transmitter errors. Refer to Background Document for guideline FR-H.1.
- Parameters (X.01) and (X.02) are described in subsection 2.2.4, Plant-Specific Symptoms for Loss of Heat Sink, of this background document. Parameters and setpoints in this CAUTION should be consistent with Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.___	**CHECK CORE COOLING:**	
	☐ a) Core Exit TCs - LESS THAN 1200°F	☐ a) GO TO Step 21.
	☐ b) At least two Hot Leg temperatures - LESS THAN 345°F	☐ b) RETURN TO Step 17.
	☐ c) RVLIS full range indication - GREATER THAN 67%	☐ c) RETURN TO Step 17.
20.___	GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, STEP 15	

NOTE: Normal conditions are desired but not required for starting the RCPs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21.___	**CHECK IF RCPs SHOULD BE STARTED:**	
	☐ a) Core exit TCs - GREATER THAN 1200°F	☐ a) GO TO Step 22.
	b) Check if an idle RCS loop is available:	b) Do the following:
	☐ • Narrow range SG level - GREATER THAN 11% [22%]	☐ 1) IF required, THEN reset both trains of Phase A Isolation.
	☐ • RCP in associated loop - AVAILABLE AND NOT OPERATING	☐ 2) IF required, THEN reset both trains of Phase B Isolation.
		☐ 3) Verify at least one Air Compressor is supplying Instrument Air System.
		☐ IF NOT, THEN start at least one Air Compressor.

(STEP 21 CONTINUED ON NEXT PAGE)

FYI, support distractor plausibility see basis next pg.

NAPS step 21

STEP: Check If RCPs Should Be Started

PURPOSE: To ensure core exit TC temperatures are greater than 1200°F before restarting RCPs

BASIS:

The operator will enter this step if:

a. He is unable to depressurize the SGs; or
b. SG depressurization was not effective in restoring adequate core cooling; or
c. Secondary heat sink is lost

The actions of Step 18 may provide temporary core cooling until some form of makeup flow to the RCS is established or one of the above items is restored.

To temporarily restore core cooling, the operator is instructed to start RCPs one at a time until core exit TCs are less than 1200°F. The RCPs should force two phase flow through the core, temporarily keeping it cool. Even single phase forced steam flow will cool the core for some time provided the RCPs can be kept running and a heat sink is available.

Starting the RCPs in this step when the core exit temperatures are greater than 1200°F will result in the clearing of the water inventory in the RCS intermediate leg (loop seal) and permit the circulation of hot gases from the overheated core to circulate through the steam generators. If the water level in the steam generators is very low at the time the RCPs are started, high steam generator tube temperatures would occur, leading to possible creep failure of the steam generator tubes. Therefore, RCPs are only started in this step if there is sufficient water level in their associated steam generator to protect the steam generator tubes from creep rupture.

If RCP restart is not effective in decreasing core exit TC temperatures below 1200°F, then the PRZR PORVs should be opened. Opening the PRZR PORVs may help reduce RCS pressure enough to cause low-head safety injection. If core exit TCs remain above 1200°F after all PRZR PORVs and block valves are open, the operator is instructed to open all other RCS vent paths to containment to reduce RCS pressure.

The pressurizer PORVs require instrument air for long-term operation, however,